UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 2, 2021

Ms. Kim Maza
Site Vice President
Shearon Harris Nuclear Power Plant
Mail Code NHP01
5413 Shearon Harris Road
New Hill, NC 27562-9300

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 184 REGARDING TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-505, REVISION 2, "PROVIDE RISK INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4B," (EPID L-2019-LLA-0218)

Dear Ms. Maza:

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 184 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment revises Technical Specifications (TSs) requirements in response to your application dated October 7, 2019, as supplemented by letters dated July 27, 2020, and November 11, 2020.

The amendment revises TSs requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met. The changes are consistent with Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's regular monthly *Federal Register* notice.

Sincerely,

/RA/

Tanya E. Hood, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 184 to NPF-63
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 184
Renewed License No. NPF-63

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated October 7, 2019, as supplemented by letters dated July 27, 2020 and November 11, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 184, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 180 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

Digitally signed by David J. Wrona
Date: 2021.04.02 10:12:25 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and Technical Specifications

Date of Issuance: April 2, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 184

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change:

Remove	Insert
Page 4	Page 4

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert	Remove	Insert
3/4 1-8	3/4 1-8	3/4 6-4a	3/4 6-4a
3/4 1-10	3/4 1-10	3/4 6-11	3/4 6-11
3/4 3-3	3/4 3-3	3/4 6-13	3/4 6-13
3/4 3-6	3/4 3-6	3/4 6-14	3/4 6-14
3/4 3-7	3/4 3-7	3/4 7-4	3/4 7-4
3/4 3-8	3/4 3-8	3/4 7-9	3/4 7-9
3/4 3-18	3/4 3-18	3/4 7-11	3/4 7-11
3/4 3-21	3/4 3-21	3/4 7-12	3/4 7-12
3/4 3-22	3/4 3-22	3/4 7-30	3/4 7-30
3/4 3-24	3/4 3-24	3/4 8-1	3/4 8-1
3/4 3-26	3/4 3-26	3/4 8-2	3/4 8-2
3/4 3-27	3/4 3-27	3/4 8-3	3/4 8-3
---	3/4 3-27a	3/4 8-4	3/4 8-4
3/4 4-11	3/4 4-11	3/4 8-12	3/4 8-12
3/4 5-3	3/4 5-3	3/4 8-17	3/4 8-17
		--	6-19k

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 184, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.
* On April 29, 2013, the name of "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc." On August 1, 2015, the name "Duke Energy Progress, Inc." was changed to "Duke Energy Progress, LLC."

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tank via a boric acid transfer pump and a charging/safety injection pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging/ safety injection pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header tank is greater than or equal to 65°F when a flow path from the boric acid tank is used;

b. At the frequencies specified in the Surveillance Frequency Control Program by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

c. At the frequencies specified in the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal; and

d. At the frequencies specified in the Surveillance Frequency Control Program by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR) at 200°F within the next 6 hours; restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

-- NOTE --

*One charging/safety injection pump train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging/safety injection pumps shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the Reactor Coolant System and reactor coolant pump seals, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to the INSERVICE TESTING PROGRAM.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
12.	Reactor Coolant Flow--Low					
	a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
	b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	13a
13.	Steam Generator Water Level--Low-Low	3/stm. gen	2/stm. gen. in any operating stm. gen.	2/stm. gen. each operating stm. gen.	1, 2	6 (1)
14.	Steam Generator Water Level--Low Coincident With Steam / Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen.	1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6
15.	Undervoltage--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

**Whenever Reactor Trip Breakers are to be tested.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) The applicable MODES for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk-Informed Completion Time Program,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, and verify compliance with the shutdown margin requirements of Specification 3.1.1.2 within 1 hour and at least once per 12 hours thereafter.

b. With no channels OPERABLE, open the Reactor Trip System breakers within 1 hour and suspend all operations involving positive reactivity changes. Verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.2 within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk-Informed Completion Time Program, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 10 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 12 - No additional corrective actions are required.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 13a - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-1	3	2	2	1, 2, 3, 4	26
d. Pressurizer Pressure--Low	3	2	2	1, 2, 3#	19
e. Steam Line Pressure--Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	19

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
(1) RCS Leak Detection (normal purge)	1	See Table 3.3-6, Item 1b1, for initiating functions and requirements.			
(2) Preentry Purge Detector	1	See Table 3.3-6, Item 1b2, for initiating functions and requirements.			
c. Airborne Particulate Radioactivity					
(1) RCS Leak Detection (normal purge)	1	See Table 3.3-6, Item 1C1, for initiating functions and requirements.			
(2) Preentry Purge Detector	1	See Table 3.3-6, Item 1C2, for initiating functions and requirements.			
5) Manual Phase "A" Isolation	See Item 3.a.1) above for Manual Phase "A" Isolation initiating functions and requirements.				
4. Main Steam Line Isolation					
a. Manual Initiation					
1) Individual MSIV Closure	1/steam line	1/steam line	1/operating steam line	1, 2, 3, 4	23
2) System	2	1	2	1, 2, 3	27

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Main Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High-2	3	2	2	1, 2, 3	26
d. Steam Line Pressure--Low	See Item 1.e. above for Steam Line Pressure--Low initiating functions and requirements.				
e. Negative Steam Line Pressure Rate--High	3/steam line	2 in any steam line	2/steam line	3***, 4***	19
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	24
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen. in each stm. gen.	1, 2	19
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
g. Steam Line Differential Pressure--High	3/steam line	2/steam line twice with any steamline low	2/steam line	1, 2, 3	26
Coincident With Main Steam Line Isolation (Causes AFW Isolation)	See Item 4. above for all Steam Line Isolation initiating functions and requirements				
7. Safety Injection Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	16
Coincident With Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Containment Spray Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

**During CORE ALTERATIONS or movement of irradiated fuel in containment, refer to Specification 3.9.9.

***Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour or in accordance with the Risk-Informed Completion Time Program.

ACTION 15a - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour. With less than the minimum channels OPERABLE, operation may proceed provided the minimum number of channels is restored within one hour, otherwise declare the affected diesel generator inoperable. When performing surveillance testing of either primary or secondary undervoltage relays, the redundant emergency bus and associated primary and secondary relays shall be OPERABLE.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition within 6 hours and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the Containment Purge Makeup and Exhaust Isolation valves are maintained closed while in MODES 1, 2, 3 and 4 (refer to Specification 3.6.1.7). For MODE 6, refer to Specification 3.9.4.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours or in accordance with the Risk-Informed Completion Time Program, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated equipment inoperable and take the appropriate ACTION required in accordance with the specific equipment specification.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 25 - During CORE ALTERATIONS or movement of irradiated fuel within containment, comply with the ACTION statement of Specification 3.9.9.

ACTION 26 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION STATEMENTS (Continued)

ACTION 27 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

   1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply ACTION b.1 or b.2, above, as appropriate, for the isolated PORV(s).

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

-- NOTE --

*One ECCS subsystem train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the following valves are in the indicated positions with the control power disconnect switch in the "OFF" position, and the valve control switch in the "PULL TO LOCK" position:

LIMITING CONDITION FOR OPERATION

b. One or more containment air locks with containment air lock interlock mechanism inoperable.##

1. Within one hour, verify an OPERABLE door is closed in the affected air lock, and
2. Within 24 hours, lock an OPERABLE door closed in the affected air lock, and
3. Once per 31 days, verify the OPERABLE door is locked closed in the affected air lock*, or
4. Otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. One or more containment air locks inoperable for reasons other than 3.6.1.3.a or 3.6.1.3.b.

1. Immediately initiate action to evaluate containment leakage rate per LCO 3.6.1.2, and
2. Within one hour, verify a door is closed in the affected air lock, and
3. Within 24 hours or in accordance with the Risk-Informed Completion Time Program, restore air lock to OPERABLE status, or
4. Otherwise be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

\## 1. ACTIONS 3.6.1.3.b.1, 3.6.1.3.b.2, 3.6.1.3.b.3, and 3.6.1.3.b.4 are not applicable if both doors in the same air lock are inoperable and ACTION 3.6.1.3.c is entered.

2. Entry and exit of containment is permissible under the control of a dedicated individual.

* Air lock doors in high radiation areas may be verified closed by administrative means.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours** or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. Refer also to Specification 3.6.2.3 Action.

-- NOTE --

**One Containment Spray System train is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position*;

b. By verifying that, on an indicated recirculation flow of at least 1832 gpm, each pump develops a differential pressure of greater than or equal to 186 psi when tested pursuant to the INSERVICE TESTING PROGRAM;

c. At the frequency specified in the Surveillance Frequency Control Program by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal and

   2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

   3. Verifying that, coincident with an indication of containment spray pump running, each automatic valve from the sump and RWST actuates to its appropriate position following an RWST Lo-Lo test signal.

d. At the frequency specified in the Surveillance Frequency Control Program by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

e. At the frequency specified in the Surveillance Frequency Control Program by verifying that containment spray locations susceptible to gas accumulation are sufficiently filled with water.

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* Not required to be met for system vent flow paths opened under administrative control.

LIMITING CONDITION FOR OPERATION

3.6.2.3 Four containment fan coolers (AH-1, AH-2, AH-3, and AH-4) shall be OPERABLE with one of two fans in each cooler capable of operation at low speed. Train SA consists of AH-2 and AH-3. Train SB consists of AH-1 and AH- 4.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers to OPERABLE status within 7 days or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required trains of fan coolers to OPERABLE status within 7 days of initial loss or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one train of the above required containment fan coolers inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable train of containment fan coolers to OPERABLE status within 7 days of initial loss or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

-- NOTE --

*One train of containment fan coolers and one Containment Spray System train are allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each train of containment fan coolers shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Starting each fan train from the control room, and verifying that each fan train operates for at least 15 minutes, and

2. Verifying a cooling water flow rate, after correction to design basis service water conditions, of greater than or equal to 1300 gpm to each cooler.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that each fan train starts automatically on a safety injection test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve specified in the Technical Specification Equipment List Program, plant procedure PLP-106, shall be OPERABLE with isolation times less than or equal to required isolation times.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk-Informed Completion Time Program, or

b. Isolate each affected penetration within 4 hours or in accordance with the Risk-Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk-Informed Completion Time Program by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency buses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

--NOTE--

LCO 3.0.4.b is not applicable.

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a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible. (NOTE: LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. Following restoration of one AFW train, all applicable LCOs apply based on the time the LCOs initially occurred.)

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Demonstrating that each motor-driven pump satisfies performance requirements by either:

a) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1514 psid at a recirculation flow of greater than or equal to 50 gpm (25 KPPH), or

b) Verifying each pump develops a differential pressure that (when temperature - compensated to 70°F) is greater than or equal to 1259 psid at a flow rate of greater than or equal to 430 gpm (215 KPPH).

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours or in accordance with the Risk-Informed Completion Time Program; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2, 3, and 4:

With one MSIV inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to the INSERVICE TESTING PROGRAM. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 or 4.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two component cooling water (CCW) pumps*, heat exchangers and essential flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water flow path OPERABLE, restore at least two flow paths to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water flow paths shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

1. Each automatic valve servicing safety-related equipment or isolating non-safety-related components actuates to its correct position on a Safety Injection test signal, and
2. Each Component Cooling Water System pump required to be OPERABLE starts automatically on a Safety Injection test signal.
3. Each automatic valve serving the gross failed fuel detector and sample system heat exchangers actuates to its correct position on a Low Surge Tank Level test signal.

* The breaker for CCW pump 1C-SAB shall not be racked into either power source (SA or SB) unless the breaker from the applicable CCW pump (1A-SA or 1B-SB) is racked out.

3/4.7.4 EMERGENCY SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent emergency service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one emergency service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

-- NOTE --

*The ‘B’ Train emergency service water loop is allowed to be inoperable for a total of 7 days to allow for maintenance on the Essential Services Chilled Water System and air handlers supported by the Essential Services Chilled Water System. Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two emergency service water loops shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

1. Each automatic valve servicing safety-related equipment or isolating non-safety portions of the system actuates to its correct position on a Safety Injection test signal, and

2. Each emergency service water pump and each emergency service water booster pump starts automatically on a Safety Injection test signal.

3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.13 At least two independent Essential Services Chilled Water System loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 7 days* or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.13 The Essential Services Chilled Water System shall be demonstrated OPERABLE by:

a. Performance of surveillances as required by the INSERVICE TESTING PROGRAM, and

b. At the frequency specified in the Surveillance Frequency Control Program by demonstrating that:

1. Non-essential portions of the system are automatically isolated upon receipt of a Safety Injection actuation signal, and
2. The system starts automatically on a Safety Injection actuation signal.

*Prior to exceeding 72 hours, the compensatory measures described in HNP LAR correspondence letter RA-19-0007 shall be implemented.

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 1457 gallons of fuel,

2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

3. A separate fuel oil transfer pump.

c. Automatic Load Sequencers for Train A and Train B.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

--NOTE--

LCO 3.0.4.b is not applicable to diesel generators.

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a. With one offsite circuit of 3.8.1.1.a inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

2. Restore the offsite circuit to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

3. Verify required feature(s) powered from the OPERABLE offsite A.C. source are OPERABLE. If required feature(s) powered from the OPERABLE offsite circuit are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 24 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

b. With one diesel generator of 3.8.1.1.b inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

*2. Within 24 hours, determine the OPERABLE diesel generator is not inoperable due to a common cause failure or perform Surveillance Requirement 4.8.1.1.2.a.4#; and

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* This ACTION is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

\# Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

3. Restore the diesel generator to OPERABLE status within 72 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. Verify required feature(s) powered from the OPERABLE diesel generator are OPERABLE. If required feature(s) powered from the OPERABLE diesel generator are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 4 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

c. With one offsite circuit and one diesel generator of 3.8.1.1 inoperable:

NOTE: Enter applicable Condition(s) and Required Action(s) of LCO 3/4.8.3, ONSITE POWER DISTRIBUTION - OPERATING, when this condition is entered with no A.C. power to one train.

1. Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Following restoration of one A.C. source (offsite circuit or diesel generator), restore the remaining inoperable A.C. source to OPERABLE status pursuant to requirements of either ACTION a or b, based on the time of initial loss of the remaining A.C. source.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

d. With two of the required offsite A.C. sources inoperable:

1. Restore one offsite circuit to OPERABLE status within 24 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

2. Verify required feature(s) are OPERABLE. If required feature(s) are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 12 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) inoperable.

3. Following restoration of one offsite A.C. source, restore the remaining offsite A.C. source in accordance with the provisions of ACTION a with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. source.

e. With two of the required diesel generators inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

#2. Restore one of the diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one diesel generator, restore the remaining diesel generator in accordance with the provisions of ACTION b with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable diesel generator.

f. With three or more of the required A.C. sources inoperable:

1. Immediately enter Technical Specification 3.0.3.

2. Following restoration of one or more A.C. sources, restore the remaining inoperable A.C. sources in accordance with the provisions of ACTION a, b, c, d and/or e as applicable with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. sources.

g. Deleted.

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

h. With one automatic load sequencer inoperable:

1. Restore the automatic load sequencer to OPERABLE status within 24 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and power availability, and

b. Demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by manually transferring the onsite Class 1E power supply from the unit auxiliary transformer to the startup auxiliary transformer.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the fuel level in the day tank,

2. Verifying the fuel level in the main fuel oil storage tank,

3. Verifying the fuel oil transfer pump can be started and transfers fuel from the storage system to the day tank,

4. Verifying the diesel generator can start** and accelerate## to synchronous speed (450 rpm) with generator steady-state voltage and frequency 6900 ± 276 volts and 60 ± 0.48 Hz,

5. Verifying the diesel generator is synchronized, gradually loaded** to an indicated 6200-6400 kW*** and operates for at least 60 minutes,

6. Verifying the pressure in at least one air start receiver to be greater than or equal to 190 psig, and

7. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

** This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable, regarding loading recommendations.

*** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing or momentary variations due to changing bus loads shall not invalidate the test.

The voltage and frequency conditions shall be met within 10 seconds or gradual acceleration to no-load conditions per vendor recommendations will be an acceptable alternative.

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Emergency Battery Bank 1A-SA and either full capacity charger, 1A-SA or 1B-SA, and,

b. 125-volt Emergency Battery Bank 1B-SB and either full capacity charger, 1A-SB or 1B-SB.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt Emergency Battery and charger shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At the frequency specified in the Surveillance Frequency Control Program and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150 x 10^{-6} ohm, and

3. The average electrolyte temperature of 10 connected cells is above 70° F.

LIMITING CONDITION FOR OPERATION

ACTION:

a. With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery within 2 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified buses shall be determined energized in the required manner at the frequency specified in the Surveillance Frequency Control Program by verifying correct breaker alignment and indicated voltage on the buses.

PROCEDURES AND PROGRAMS (Continued)

r. Risk-Informed Completion Time Program

This program provides controls to calculate a Risk-Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, “Risk-Managed Technical Specifications (RMTS) Guidelines.” The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1 and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

    1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

    2. For emergent conditions, the revised RICT must be determined within the time limits of the Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

    3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

    1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

    2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 184

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application dated October 7, 2019,[1] as supplemented by letters dated July 27, 2020,[2] and November 11, 2020,[3] Duke Energy Progress, LLC (the licensee) submitted a license amendment request (LAR) for the Shearon Harris Nuclear Power Plant (Harris), Unit 1. The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018.[4]

The U.S. Nuclear Regulatory Commission (NRC or the Commission) issued a final model safety evaluation (SE) approving TSTF-505, Revision 2, on November 21, 2018.[5] The licensee has proposed variations from the TS changes described in TSTF-505, Revision 2. The variations are described in Section 2.2.3 of this SE.

The supplements dated July 27, 2020, and November 11, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 14, 2020 (85 FR 2160).

The NRC staff participated in a remote regulatory audit through an internet portal established by the licensee. The audit was conducted from June 22, 2020, to June 25, 2020. The NRC staff performed the audit to ascertain the information needed to support its review of the application and develop requests for additional information (RAIs), as needed. On December 7, 2020, the NRC staff issued an audit summary.[6]

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[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML19280C844
[2] ADAMS Accession No. ML20209A304
[3] ADAMS Accession No. ML20316A007
[4] ADAMS Accession No. ML18183A493
[5] ADAMS Accession No. ML18269A041
[6] ADAMS Accession No. ML20329A381

2.0 REGULATORY EVALUATION

2.1 DESCRIPTION OF RISK-INFORMED COMPLETION TIME PROGRAM

The TS LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, the licensee must shut down the reactor or follow any remedial or required action (e.g., testing, maintenance, or repair activity) permitted by the TSs until the condition can be met. The remedial actions (i.e., ACTIONS) associated with an LCO contain Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and completion times (CTs). The CTs are referred to as the “front stops” in the context of this SE. For certain Conditions, the TS require exiting the Mode of Applicability of an LCO (i.e., shutdown the reactor).

The licensee’s TS are not presented in the Improved Standard Technical Specifications format. The term “Action Statement” is conventionally used to describe ways in which the requirements of the LCO can fail to be met (i.e., Condition) and the necessary Required Actions. Throughout this SE, the terms “Condition” and “Required Actions” are used to describe Action Statements. The term “Allowed Outage Time” is conventionally used to describe the length of time that equipment is permitted to be inoperable. For the purposes of this SE, the terms “CT” and “Allowed Outage Time” are used interchangeably.

The Nuclear Energy Institute (NEI) Topical Report 06-09-A,[7] “Risk Informed Technical Specifications Initiative 4b: Risk Managed Technical Specification (RMTS),” Revision 0-A, dated October 2012, provides a methodology for extending existing CTs and thereby delay exiting the operational mode of applicability or taking Required Actions if risk is assessed and managed within the limits and programmatic requirements established by a RICT Program.

2.2 DESCRIPTION OF TS CHANGES

The licensee’s submittal requested approval to add a RICT Program to the Administrative Controls section of the TS, add new Action Statements in some TSs, and modify selected CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09-A. The licensee’s application for the changes proposed to use NEI 06-09-A and included documentation regarding the technical adequacy of the probabilistic risk assessment (PRA) models for the RICT Program, consistent with the guidance of Regulatory Guide (RG) 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009.[8]

2.2.1 Technical Specification 6.8.4.r Risk-Informed Completion Time Program

Technical Specification 6.8.4.r, which describes the RICT Program, would be added to the TS and reads as follows:

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[7] ADAMS Accession No. ML122860402
[8] ADAMS Accession No. ML090410014

Risk-Informed Completion Time Program

This program provides controls to calculate a Risk-Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09-A, Revision 0, “Risk-Managed Technical Specifications (RMTS) Guidelines.” The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1, and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09-A, Appendix A, must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in

Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the Completion Times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

2.2.2 Application of the RICT Program to Existing LCOs and Action Statements

The typical CT is modified by the application of the RICT Program as shown in the following example. The changed portion is indicated in italics.

ACTION:

a. With one subsystem inoperable, restore subsystem to OPERABLE status within 7 days *or in accordance with the Risk-Informed Completion Time Program.*

Where necessary, conforming changes are made to CTs to make them accurate following use of a RICT. For example, most TSs have requirements to close/isolate containment isolation devices if one or more containment penetrations have inoperable devices. This is followed by a requirement to periodically verify the penetration is isolated. By adding the flexibility to use a RICT to determine a time to isolate the penetration, the periodic verifications must then be based on the time "following isolation."

Individual LCO Action Statements and CTs modified by the proposed change are identified below.

LCO 3.3.1 Reactor Trip System Instrumentation

Action 1	With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours *or in accordance with the Risk-Informed Completion Time Program* or be in HOT STANDBY…
Action 2a.	The inoperable channel is placed in the tripped condition within 6 hours *or in accordance with the Risk-Informed Completion Time Program*.
Action 6a.	The inoperable channel is placed in the tripped condition within 6 hours *or in accordance with the Risk-Informed Completion Time Program*, and…
Action 11	With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours *or in accordance with the Risk-Informed Completion Time Program* or declare the breaker inoperable and apply ACTION 8…
Action 13	With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…

LCO 3.3.2 Engineered Safety Features Actuation System Instrumentation

Action 14	With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action 15	With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour *or in accordance with the Risk-Informed Completion Time Program*.
Action 18	With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action 19a.	The inoperable channel is placed in the tripped condition within 6 hours *or in accordance with the Risk-Informed Completion Time Program*, and…
Action 21	With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action 24	With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…

LCO 3.4.4 Relief Valves

Action b.1.	With only one safety grade PORV [Power Operated Relief Valve] OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…

LCO 3.5.2 ECCS [Emergency Core Cooling Systems] Subsystems – T_{avg} [average coolant temperature] Greater Than or Equal to 350°F [degrees Fahrenheit]

Action a.	With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours* *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.6.1.3 Containment Air Locks

Action c.3.	Within 24 hours *or in accordance with the Risk-Informed Completion Time Program*, restore air lock to OPERABLE status, or…

LCO 3.6.2.1 Containment Spray System

Action	With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours** *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…

LCO 3.6.2.3 Containment Cooling System

Action a.	With one train of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore the inoperable train of fan coolers to OPERABLE status within 7 days *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...
Action b.	With both trains of the above required containment fan coolers inoperable and both Containment Spray Systems OPERABLE, restore at least one train of fan coolers to OPERABLE status within 72 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY... Restore both above required trains of fan coolers to OPERABLE status within 7 days of initial loss *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action c.	With one train of the above required containment fan coolers inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours* *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY… Restore the inoperable train of containment fan coolers to OPERABLE status within 7 days of initial loss *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.6.3 Containment Isolation Valves

Action b.	Isolate each affected penetration within 4 hours *or in accordance with the Risk-Informed Completion Time Program* by use of at least one deactivated automatic valve secured in the isolation position, or
Action c.	Isolate each affected penetration within 4 hours *or in accordance with the Risk-Informed Completion Time Program* by use of at least one closed manual valve or blind flange, or…

LCO 3.7.1.2 Auxiliary Feedwater System

Action a.	With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours* *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.7.1.5 Main Steam Line Isolation Valves

Action	MODE 1: With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours *or in accordance with the Risk-Informed Completion Time Program*, otherwise be in HOT STANDBY…

LCO 3.7.3 Component Cooling Water System

Action	With only one component cooling water flow path OPERABLE, restore at least two flow paths to OPERABLE status within 72 hours** *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.7.4 Emergency Service Water System

Action	With only one emergency service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours* *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.8.1.1 A.C. Sources Operating

Action a.2.	Restore the offsite circuit to OPERABLE status within 72 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action b.3	Restore the diesel generator to OPERABLE status within 72 hours** *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action c.1.	Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...
Action d.1.	Restore one offsite circuit to OPERABLE status within 24 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action h.1.	Restore the automatic load sequencer to OPERABLE status within 24 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.8.2.1 D.C. Sources Operating

Action	With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.8.3.1 Onsite Power Distribution Operating

Action a.	With one of the required divisions of A.C. ESF buses not fully energized, reenergize the division within 8 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…
Action b.	With one 118-volt A.C. vital bus not energized from its associated inverter, reenergize the 118-volt A.C. vital bus within 2 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...
Action d.	With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery, reenergize the D.C. bus from its associated Emergency Battery within 2 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

2.2.3 Variations from TSTF-505, Revision 2

2.2.3.1 Application of the RICT to Additional ACTIONS Statements

The following individual LCO Actions statements and CTs identified below are modified by the proposed change to permit the application of a RICT and are in addition to those included in TSTF-505.

LCO 3.1.2.2 Flow Paths - Operating

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…

LCO 3.1.2.4 Charging Pumps – Operating

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours* *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY…

LCO 3.3.2 Engineered Safety Features Actuation System Instrumentation

ACTION:

14. (Application of this Action to FUNCTIONAL UNIT 8.a Containment Spray Switch over to Containment Sump Automatic Actuation Logic and Actuation Relays) With the number of OPERABLE channels one less

than the Minimum Channels OPERABLE requirement, *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours…

LCO 3.6.3 Containment Isolation Valves

ACTION:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours *or in accordance with the Risk-Informed Completion Time Program*, or…

LCO 3.7.13 Essential Services Chilled Water System

ACTION:

With only one Essential Services Chilled Water System loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours* *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

LCO 3.8.3.1 Onsite Power Distribution Operating

ACTION:

c. With one 118-volt A.C. vital bus not energized from its associated inverter connected to its associated D.C. bus, re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours *or in accordance with the Risk-Informed Completion Time Program* or be in at least HOT STANDBY...

2.2.3.2 Additional Variations from TSTF-505, Revision 2

The following individual LCO Actions statements and CTs identified below are added by the proposed change and permit the application of a RICT.

LCO 3.3.2 Engineered Safety Features Actuation System Instrumentation

ACTION:

27. With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

The following individual LCO Actions statements and CTs identified below are added by the proposed change and do not permit the application of a RICT.

LCO 3.3.1 Reactor Trip System Instrumentation

ACTION:

13a. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

LCO 3.3.2 Engineered Safety Features Actuation System Instrumentation

ACTION:

26. With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

2.3 REGULATORY REVIEW

2.3.1 Applicable Regulations

In accordance with Section 50.90, "Application for amendment of license, construction permit, or early site permit," of Title 10 of the *Code of Federal Regulations* (10 CFR), whenever a holder of a license wishes to amend the license, including TSs in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate.

The regulation at 10 CFR 50.36(c)(2) requires that TSs contain LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor, or follow any remedial action permitted by the TSs until the LCO can be met. Typically, the TSs require restoration of equipment in a timeframe commensurate with its safety significance, along with other engineering considerations. The regulation at 10 CFR 50.36(b) requires that TSs be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto.

In determining whether the proposed TS remedial actions should be granted, the Commission will apply the "reasonable assurance" standards of 10 CFR 50.40(a) and 50.57(a)(3). The regulation at 10 CFR 50.40(a) states that in determining whether to grant the licensing request, the Commission will be guided by, among other things, consideration about whether "the processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals, in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20 of this chapter, and that the health and safety of the public will not be endangered."

The regulation at 10 CFR 50.36(c)(5) states that administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

The regulation at 10 CFR 50.55a(h) "Protection and safety systems" states, in part, that protection systems of nuclear power reactors of all types must meet the requirements specified in this paragraph.

Section 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), requires licensees to monitor the performance or condition of SSCs against licensee established goals in a manner sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. The regulation at 10 CFR 50.65(a)(4) requires the assessment and management of the increase in risk that may result from a proposed maintenance activity.

2.3.2 Commission Policy

The NRC provided details concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," published in the *Federal Register* (60 FR 42622; August 16, 1995). In this publication, the Commission wrote, in part:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach….
>
> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner….

> Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data….
>
> Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:
>
> (1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.
>
> (2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.
>
> (3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.
>
> (4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgements on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

2.3.3 Regulatory Guidance

Revision 3 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018,[9] describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent

[9] ADAMS Accession No. ML17317A256

licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

Revision 1 of RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," May 2011,[10] describes an acceptable risk-informed approach specifically for assessing proposed TS changes. This RG identifies a three-tiered approach for a licensee's evaluation of the risk associated with a proposed TS CT change, as follows.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on plant risk as expressed by the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). The limits for ICCDP and ICLERP are consistent with the criteria for incremental core damage probability (ICDP) and incremental large early release probability (ILERP) from the Nuclear Management and Resources Council (NUMARC) 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 2011,[11] guidance for managing the risk of on-line maintenance activities. The ICDP and ILERP are the limits on which licensee will base the RICT. This guidance was endorsed by the NRC staff in RG 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," May 2012,[12] for compliance with the Maintenance Rule, 10 CFR 50.65(a)(4). Tier 1 also addresses PRA acceptability, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the proposed TS change is considered with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is removed from service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee's Configuration Risk Management Program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk-significant configurations that may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule, which requires a licensee to assess

[10] ADAMS Accession No. ML100910008
[11] ADAMS Accession No. ML11116A198
[12] ADAMS Accession No. ML113610098

and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventative maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP ensures that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

Revision 2 of RG 1.200 describes an acceptable approach for determining whether the PRA acceptability, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. This RG provides guidance for assessing the technical adequacy of a PRA. Revision 2 of RG 1.200, endorses, with clarifications and qualifications, the use of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard, RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (i.e., the PRA Standard).

As discussed in RG 1.177, Revision 1, and RG 1.174, Revision 3, a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption;

2. The proposed change is consistent with the defense-in-depth philosophy;

3. The proposed change maintains sufficient safety margins;

4. When proposed changes result in an increase in core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement; and

5. The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-505, Revision 2, provides for the addition of a RICT Program to the Administrative Controls section of the TS and modifies selected Required Action CTs to permit extending the CTs, provided risk is assessed and managed as described in NEI 06-09-A. In accordance with NEI 06-09-A, PRA is used to justify each extension to a Required Action CT based on the specific plant configuration which exists at the time of the applicability of the Required Action and are updated when plant conditions change. The licensee's LAR included documentation regarding the technical adequacy of the PRA models used in the CRMP also known as Real-Time Risk (RTR) consistent with the guidance of RG 1.200.

Most TS identify one or more Conditions for which the LCO may not be met, to permit a licensee to perform required testing, maintenance, or repair activities. Each Condition has an associated Required Action for restoration of the LCO or for other actions, each with some fixed time interval, referred to as the CT, which identifies the time interval permitted to complete the Required Action. Upon expiration of the CT, the licensee is required to shut down the reactor, or follow the Required Action(s) stated in the ACTIONS requirements. The RICT Program provides the necessary administrative controls to permit extension of CTs and thereby delay reactor shutdown or Required Actions, if risk is assessed and managed within specified limits and programmatic requirements. The specified safety function or performance level of TS required equipment is unchanged, and the Required Action(s), including the requirement to shut down the reactor are also unchanged, only the CTs for the Required Actions are extended by the RICT Program.

3.1 REVIEW OF KEY PRINCIPLES

Revision 1 of RG 1.177 and RG 1.174, Revision 3, identify five key safety principles to be applied to risk-informed changes to the TSs. Each of these principles are addressed in NEI 06-09-A. The NRC staff's evaluation of the licensee's proposed use of RICTs against these key safety principles is discussed below.

3.1.1 Key Principle 1: Evaluation of Compliance with Current Regulations

As stated in 10 CFR 50.36(c)(2):

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

When the necessary redundancy is not maintained (e.g., one train of a two-train system is inoperable), the TSs permit a limited period of time to restore the inoperable train to operable status and/or take other remedial measures. If these actions are not completed within the CT, the TSs normally require that the plant exit the mode of applicability for the LCO. With one train of a two-train system inoperable, the TS safety function is accomplished by the remaining operable train. In the current TSs, the CT is specified as a fixed time period (termed the "front stop"). The addition of the option to determine the CT in accordance with the RICT Program would allow an evaluation to determine a configuration-specific CT. The evaluation would be done in accordance with the methodology prescribed in NEI 06-09-A and TS 6.8.4.r. The RICT is limited to a maximum of 30 days (termed the "back stop"). The CTs in the current TSs were established using experiential data, risk insights, and engineering judgement. The RICT Program provides the necessary administrative controls to permit extension of CTs and thereby delay reactor shutdown or Required Actions, if risk is assessed and managed appropriately within specified limits and programmatic requirements.

When the necessary redundancy is not maintained, and the system loses the capability to perform its safety function(s) without any further failures (e.g., two trains of a two-train system are inoperable), the plant must exit the mode of applicability for the LCO, or take remedial actions, as specified in the TSs. A configuration-specific RICT may not be used in this condition. With the incorporation of the RICT Program, the required performance levels of equipment specified in LCOs are not changed. Only the required CTs for the Required Actions are modified by the RICT Program.

3.1.1.1 Key Principle 1 Conclusions

Based on the discussion provided above, the NRC staff finds that the proposed changes meet the first key safety principle of RG 1.174, Revision 3, and RG 1.177, Revision 1.

3.1.2 Key Principle 2: Evaluation of Defense-in-Depth

Defense-in-depth is an approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

As discussed throughout RG 1.174, consistency with the defense-in-depth philosophy is maintained by the following measures:

- Preserve a reasonable balance among the layers of defense.

- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.

- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.

- Preserve adequate defense against potential CCFs.

- Maintain multiple fission product barriers.

- Preserve sufficient defense against human errors.

- Continue to meet the intent of the plant’s design criteria.

The proposed change represents a robust technical approach that preserves a reasonable balance among redundant and diverse key safety function that provide avoidance of core damage, avoidance of containment failure, and consequence mitigation. The three-tiered approach to risk-informed TS CT changes provides additional assurance that defense-in-depth will not be significantly impacted by such changes to the licensing basis. The licensee is proposing no changes to the design of the plant or any operating parameter, no new operating configurations, and no new changes to the design basis in the proposed changes to the TS.

The effect of the proposed changes when implemented will be that the RICT Program will allow CTs to vary based on the risk significance of the given plant configuration (i.e., the equipment out-of-service at any given time) provided that the system(s) retain(s) the capability to perform the applicable safety function(s) without any further failures (e.g., one train of a two-train system is inoperable). A configuration-specific RICT may not be used if the system has lost the capability to perform its safety function(s). These restrictions on inoperability of all required trains of a system ensure that consistency with the defense-in-depth philosophy is maintained by following existing guidance when the capability to perform TS safety function(s) is lost.

The proposed RICT Program uses plant-specific operating experience for component reliability and availability data. Thus, the allowances permitted by the RICT Program are directly reflective of actual component performance in conjunction with component risk significance. In some cases, the RICT Program may use compensatory actions to reduce calculated risk in some configurations. Where credited in the PRA, these actions are incorporated into station procedures or work instructions and have been modeled using appropriate human reliability considerations. Application of the RICT Program determines the risk significance of plant configurations. It also permits the operator to identify the equipment that has the greatest effect on the existing configuration risk. With this information, the operator can manage the out-of-service duration and determine the consequences of removing additional equipment from service.

The application of the RICT Program places high value on key safety functions and works to ensure they remain a top priority over all plant conditions. The RICT will be applied to extend CTs on key electrical power distribution systems. Failures in electrical power distribution systems can simultaneously affect multiple safety functions; therefore, potential degradation to defense-in-depth during the extended CTs is discussed further below.

3.1.2.1 Use of Compensatory Measures to Retain Defense-in-Depth

Application of the RICT Program provides a structure to assist the operator in identifying effective compensatory actions for various plant maintenance configurations to maintain and manage acceptable risk levels. Topical Report NEI 06-09-A addresses potential compensatory actions and RMA measures by stating, in generic terms, that compensatory measures may include but are not limited to the following:

- Reduce the duration of risk-sensitive activities.
- Remove risk-sensitive activities from the planned work scope.
- Reschedule work activities to avoid high risk-sensitive equipment outages or maintenance states that result in high-risk plant configurations.
- Accelerate the restoration of out-of-service equipment.
- Determine and establish the safest plant configuration.

Topical Report NEI 06-09-A requires that compensatory measures be initiated when the PRA calculated RMA time is exceeded, or for preplanned maintenance for which the RMA time is expected to be exceeded, RMAs shall be implemented at the earliest appropriate time.

3.1.2.2 Evaluation of Electrical Power Systems

According to the Harris Updated Final Safety Analysis Report (UFSAR), the plant is designed such that the safety functions are maintained assuming a single failure within the electrical power system. By incorporating an electrical power supply perspective, this concept is further reflected in a number of principal design criteria. Single-failure requirements are typically suspended for the time that a plant is not meeting an LCO (i.e., in an ACTION statement). This section considers the plant configurations from a defense-in-depth perspective.

3.1.2.2.1 Electrical Power Systems Description

According to Harris UFSAR Chapter 8, “Electric Power,” the offsite power system includes eight 230 kilovolts (kV) transmission lines that connect the switchyard to the transmission network. The power from the eight transmission lines is provided to two startup transformers through a breaker-and-a-half scheme arrangement. These two startup transformers have sufficient capacity to provide for start-up and full load operation of the unit and are the preferred sources of offsite power for start-up and shutdown, when the main generator is unavailable. Each startup transformer has sufficient capacity to support and maintain plant shutdown. An additional path of power supply from the high voltage grid to the Plant Electric Power Distribution System can be made available after disconnecting the main generator from the 22 kV bus. The 22 kV bus power supply can then be rearranged from its normal configuration into a configuration such that power can be fed from the offsite power system through the main transformer bank to the unit auxiliary transformer, leaving the main generator disconnected. This last path for providing power through main and auxiliary transformer is not considered for estimating the RICTs. Two independent offsite sources from the two start up transformers are considered for the discussion related to RICT and the Action statements when the generator is assumed offline.

The onsite power system includes two 6.9 kV Engineered Safety Features (ESF) buses (1A-SA and 1B-SB), two emergency diesel generators (EDGs), several 480 volts (V) buses supplying loads either directly or through motor control centers (MCCs), two 125V direct current (DC) ESF batteries (1A-SA and 1B-SB), four 120V alternating current (AC) ESF uninterruptible buses, two 125V DC ESF buses, and several ESF 208V/120V power distribution panels. The automatic start and initiation circuits for the EDGs are provided through the Engineered Safety Features Actuation System (ESFAS) or by the under-voltage relay. The EDGs are air started and water cooled. There are two tanks of compressed air per EDG, each with capability of five starts. As described in Section 9.5.5 of the UFSAR, the closed-loop cooling system with the tube side of the heat exchanger is supplied with cooling water from the Emergency Service Water (ESW) system. Upon loss of offsite power (LOOP), the ESW pump will start and run with the power provided from the associated EDG. The ESW pumps will supply cooling water to the EDG after a period of 20-25 seconds (an EDG can run one minute without cooling). The EDG jacket cooling from ESW is arranged as a split system; header A is dedicated EDG 1A-SA and header B is dedicated EDG 1B-SB. As described in Section 9.5.4 of the UFSAR, the day tank for each EDG contains 3000 gallons of fuel sufficient for approximately 7 hours of run time (or 445 gallons per hour). Each fuel storage tank has enough capacity (175,000 gallons and is refilled at 88,000 gallons) for at least 7 days of EDG operation. There are two fuel oil transfer pumps with a capacity of 40 gallons per minute. One fuel oil transfer pump is required to transfer sufficient fuel oil supply for each EDG.

The DC power is provided by emergency DC systems for excitation, an EDG control panel and an engine control panel. The DC support system is divisional with DC Train A feeding EDG 1A-SA and DC Train B feeding EDG 1B-SB. There are two trains of DC power, each powered by a bank of 125V batteries. These safety-related batteries are required to provide DC power for two hours following design basis events. As described in Section 8.3.1.2.21 of the UFSAR, Harris is subject to a minimum station blackout coping capability of four hours. Each DC bus is equipped with two battery chargers which will be energized from emergency AC (if available) within one minute. There is a third 125V battery, designated 1A, that feeds the non-safety related switchgear, emergency control room lighting and other 125V DC non-safety loads. This would allow recovery and reconnection of offsite power after the safety batteries are depleted. There is also a 250V station battery that feeds the non-safety related 250V loads.

The NRC staff requested additional information from the licensee in an email dated September 29, 2020.[13] In RAI 7, the NRC staff asked the licensee to clarify that, without the batteries, the battery chargers alone can support either LOOP, Loss of Coolant Accident (LOCA), or LOOP-and-LOCA initiators.

In its RAI response dated November 11, 2020, the licensee stated that the chargers will be in current limit operation mode (115% of rated) for the LOCA, LOOP, and LOCA coincident with LOOP (LOCA/LOOP) conditions. The chargers enter the current limit condition during Load Block 1 & 2 in the first 0-10 seconds of the sequencer actuation due to the EDG field flashing and switchgear breaker trip coils all operating at once (LOCA/LOOP and LOOP), followed by 20 Kilovolt-Ampere (KVA) in subsequent load blocks. The chargers therefore are not credited for LOOP, LOCA, and LOCA/LOOP conditions as independent and redundant sources of DC power. For all other transients, the chargers are capable of carrying the required loads and are considered as independent and redundant sources for DC power.

As described in Section 8.3.1.1.1.4 of the UFSAR, Harris has four uninterruptible buses (Channels I, II, III, and IV), each of which is fed by two sources. Each safety-related 7.5 KVA inverter is nominally rated at 118 VAC, 60 Hertz (Hz), normally supplied through its rectifier from a 480V ESF MCC. Should the voltage drop below the required level, the power for the inverter is supplied automatically from a 125V DC ESF battery. Blocking diodes in each input circuit prevent voltage feedback. Harris does not have any swing bus. For systems with more than two divisions, the third division can be connected manually to either Division A or Division B. For example, to facilitate the reconnection of the power feeders for component cooling water pump “C”, two separate independent feeder circuits, one from each division, are routed to separate locations close to the motor. However, only one feeder circuit breaker is provided for each third service motor. Reconnection from one division to another requires that the breaker be physically removed from one bus and that same breaker be installed in the other bus.

As described in Section 8.3.1.1.1.6 of the UFSAR, Harris is also equipped with Dedicated Shutdown Diesel Generator (DSDG). The DSDG is an independent, outdoor, non-safety related diesel generator rated at 400 kilowatts, 0.8 power factor, 480 VAC, and is equipped with its accessories and fuel storage system. It is a standby generator with average power output corresponding to 70% of the standby power rating. The DSDG provides back-up power for 480V MCC 1D23 upon loss of normal source (bus 1D2) via an automatic transfer switch and can supply all loads fed by the MCC. MCC 1D23 provides back-up power to the safety-related battery chargers, which is normally fed by bus 1D2 (with a back-up source of the DSDG). A separate feed is provided for each train of the chargers. The DSDG is designed to fast auto-start by a DC electric starter motor which is supplied by a skid-mounted 24V battery (independent of the plant emergency DC), and it can accept up to 100% of the rated load in one step.

To support the electrical evaluation of the proposed RICT for electrical power system TS conditions in the application and to verify that the capability of the affected electrical power systems to perform their safety functions (assuming no additional failures) is maintained, the NRC staff requested additional information from the licensee in an email dated September 29, 2020. In its RAI response dated November 11, 2020, the licensee stated the DSDG is not seismically qualified. The DSDG provides an additional layer of protection (i.e., defense-in-depth) beyond the mitigation capability of the TS required EDGs, for the loss of offsite power events. Although the safety benefit of the DSDG at Harris for postulated seismic events is not

[13] ADAMS Accession No. ML20274A046

quantitatively credited, a reduction in overall seismic CDF and LERF would result if the DSDG were to be credited. The DSDG has the capability to meet the required 24-hour mission time. The DSDG fuel tank has sufficient capacity at 75% level (i.e., minimum fuel tank level provided by procedure) to meet the 24-hour mission time. The NRC staff notes that operation of 70% capacity is sufficient for DSDG. Therefore, the use of 75% capacity for estimating the fuel consumption is conservative. The DSDG is quantitatively credited as a mitigating system for non-seismic LOOP initiators.

3.1.2.2.2 Technical Evaluation

The licensee requested to use the RICT Program to extend the existing CT for the following TS 3/4.8, “Electrical Power Systems,” Actions. The NRC staff’s evaluation of the proposed changes considered a number of potential plant conditions allowed by the proposed RICTs. The NRC staff also considered the available redundant or diverse means to respond to various plant conditions. In these evaluations, the NRC staff examined the safety significance of different plant conditions resulting in both shorter and longer CTs. The plant conditions evaluated are discussed in more detail below.

The NRC staff reviewed the information pertaining to the proposed electrical power systems TS Actions in the application, the UFSAR, and applicable TS LCO and TS Bases to verify the capability of the affected electrical power systems to perform their safety functions (assuming no additional failures) is maintained. To achieve that objective, the NRC staff verified whether each proposed TS Action’s design success criteria reflect the redundant or absolute minimum electrical power source/subsystem required to be operable by the LCOs to support the safety functions necessary to mitigate postulated design bases accidents (DBA), safely shutdown the reactor, and maintain the reactor in a safe shutdown condition. The NRC staff further reviewed the remaining credited power source/equipment to verify whether the proposed TS Action satisfies its design success criteria. In conjunction with reviewing the remaining credited power source/equipment, the NRC staff considered supplemental electrical power sources/equipment (not necessarily required by the LCOs and which can be either safety or non-safety related) that are/is available at Harris and capable of performing the same safety function of the inoperable electrical power source/equipment. In addition, the NRC staff reviewed the proposed RMA examples for reasonable assurance that these RMAs are appropriate to monitor and control risk for applicable (e.g., monitoring and implementing actions to prevent the remaining components and systems from becoming unavailable due to routine preventive maintenance) for applicable TS Actions. The changed portion is indicated in italics.

Enclosure 12 of the LAR describes the process for identification of RMAs applicable during extended CTs and provides examples of RMAs. It is stated in the LAR that RMAs will be governed by plant procedures for planning and scheduling maintenance activities. The procedures provide guidance for the determination and implementation of RMAs when entering the RICT Program consistent with the guidance provided in NEI 06-09-A. As a result of the audit conducted by the NRC staff to ascertain the information needed to support its review of the application, the licensee further provided RMA examples in the supplement dated July 27, 2020.

LCO 3.8.1.1 A.C. Sources Operating - ACTION a - One required offsite circuit inoperable

Current Required Action a2:

> Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Proposed Required Action a2:

> Restore the offsite circuit to OPERABLE status within 72 hours *or in accordance with Risk-Informed Completion Time Program,* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

The licensee has requested to use the RICT Program for TS 3.8.1.1 Action a2. For this TS Action, per the LAR, an estimated RICT of 8.9 days is calculated. In the supplement dated July 27, 2020, the licensee stated that the approximate range of 2.3 to 8.9 days is determined for RICT depending on the possible emergent conditions. Under normal operating conditions, the plant electric power distribution system receives power from the main generator through two-unit auxiliary transformers. During normal operation when the generator is online, the power can also be provided by the main generator through the auxiliary transformer. For start-up and shutdown conditions, when the main generator is unavailable, power is obtained through two SATs from the grid through the 230 kV switchyard.

The NRC staff notes that with one offsite circuit inoperable, power required for the plant shutdown can be provided through three redundant and diverse means: either by the remaining offsite circuit or by either one of the two EDGs. In addition, the non-safety DSDG, a supplemental power source, can provide the required power to maintain the plant in hot shutdown. The most limiting accident condition assumed in Chapter 15 of the Harris UFSAR, is the LOCA/LOOP. The design success criterion for this case is one of two EDGs feeding the emergency bus with all associated equipment of the emergency bus operating. For LOOP scenarios not coincident with LOCA, the DSDG and one of two EDGs feeding the emergency bus can meet the design success criterion.

The NRC staff finds that the design success criterion will be met during the entry of the RICT Program for TS 3.8.1.1 Action a2. The NRC staff examined the RMA examples for TS 3.8.1.1 Action a and found that these RMAs are reasonable and consistent with the intent of NEI 06-09 Section 3.4.3 for risk awareness and control during the emergent scenarios for TS 3.8.1.1 Action a. Considering that the design success criterion will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.1.1 Action a2, the NRC staff finds that the proposed changes to TS 3.8.1.1 Action a2 is acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the AC sources to perform their safety functions (assuming no additional failures of electrical components) is maintained.

LCO 3.8.1.1 A.C. Sources Operating - ACTION b - One required diesel generator inoperable

Current Required Action b3

> Restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Proposed Required Action b3:

> Restore the diesel generator to OPERABLE status within 72 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

The licensee has requested to use the RICT Program for TS 3.8.1.1 Action b3. For this TS Action, per the LAR, an estimated RICT of 26 days is calculated. The licensee stated in its supplement dated July 27, 2020, that the approximate range determined for the RICT depending on possible emergent conditions is estimated from 3.7 days to 26.6 days.

The NRC staff notes that with one required EDG inoperable, power can be provided through three redundant and diverse means; either by the remaining operable EDG, or by either one of the two independent offsite circuits. While in this TS Action, the remaining EDG is tested to verify that CCF has not occurred. In addition, the non-safety DSDG, a supplemental power source, can provide the required power to maintain the plant in hot shutdown. The most limiting accident condition is the LOCA/LOOP. The design success criterion for this case is one EDG feeding the emergency bus with all associated equipment of the emergency bus operating. For this case, the redundancy is reduced to one remaining EDG. Even though there is only a single layer of defense-in-depth for the TS 3.8.1.1 Action b during the worst case scenario, by considering the low probability of LOCA coincident with LOOP and the implementation of RMAs during the RICT Program entry for TS 3.8.1.1 Action b3, the NRC staff finds it reasonable and acceptable.

The NRC staff finds that the design success criterion will be met during the entry of the RICT Program for TS 3.8.1.1 Action b3. The NRC staff examined the RMA examples for TS 3.8.1.1 Action b and found that these RMAs are reasonable and consistent with the intent of NEI 06-09 Section 3.4.3 for risk awareness and control during the emergent scenarios for TS 3.8.1.1 Action b.

Considering that the design success criterion will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.1.1 Action b3, the NRC staff finds that the proposed changes to TS 3.8.1.1 Action b3 are acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the AC sources to perform their safety functions (assuming no additional failures of electrical components) is maintained.

<u>LCO 3.8.1.1 A.C. Sources Operating - ACTION c - One offsite circuit and one diesel generator inoperable</u>

Current Required Action c1

> Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Proposed Required Action c1

> Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The licensee has proposed the use of RICT Program for this TS Action, but by the licensee's current calculation, the use of the RICT Program on this TS Action is precluded by the instantaneous CDF or LERF limits of 1E-03 or 1E-04, respectively. The licensee has requested that this TS Action remain within the scope of the LAR, and proposed that the RICT Program be used on this TS Action should the plant risk estimates decrease in the future for TS 3.8.1.1 Action c1.

The NRC staff notes that with one required EDG and one offsite circuit inoperable, power can be provided through two diverse means: either by the remaining operable EDG or by the remaining offsite circuit. In addition, the non-safety DSDG, a supplemental power source, can supply the required power to maintain the plant in hot shutdown condition.

The most limiting accident condition is the LOCA/LOOP. The design success criterion for this case is the remaining EDG feeding the emergency bus with all associated equipment of the emergency bus operating. For this case, the redundancy is reduced to one remaining EDG. Even though there is only a single layer of defense-in-depth for the TS 3.8.1.1 Action c worst scenario, by considering the low probability of LOCA coincident with LOOP and the implementation of RMAs during the RICT Program entry for TS 3.8.1.1 Action c1, the NRC staff finds that this is reasonable and acceptable.

The NRC staff finds that the design success criterion will be met during the entry of the RICT Program for TS 3.8.1.1 Action c1. The NRC staff noted that the licensee may implement the RICT Program on this TS Action if the instantaneous CDF or LERF limits of 1E-03 or 1E-04, respectively are met. Considering that the design success criterion will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.1.1 Action c1, the NRC staff finds that, if the instantaneous CDF or LERF limits of 1E-03 or 1E-04 respectively are met, the proposed changes to TS 3.8.1.1 Action c1 are acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the AC sources to perform their safety functions (assuming no additional failures of electrical components) is maintained.

<u>LCO 3.8.1.1 A.C. Sources Operating - ACTION d - Two offsite circuits are inoperable</u>

Current Required Action d1

> Restore one offsite circuit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Proposed Required Action d1

> Restore one offsite circuit to OPERABLE status within 24 hours *or in accordance with Risk-Informed Completion Time Program*, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

The licensee has requested to use the RICT Program for TS 3.8.1.1 Action d1. For TS 3.8.1.1 Action d1, per the LAR, an estimated RICT of 4.8 days is calculated. The licensee stated in its supplement dated July 27, 2020, that the approximate range determined for the RICT depending on possible emergent conditions is estimated from 3.7 days to 26.6 days.

According to the Harris UFSAR, two independent 230 kV sources of offsite power are available to Harris Unit 1. During TS 3.8.1.1 Action d when the plant is in power operation, the two 6.9 kv emergency buses can be fed through the auxiliary transformers via auxiliary buses 1D and 1E.

The NRC staff notes that with both required offsite circuits inoperable and a plant trip (with turbine tripped and reactor in standby or hot shutdown mode), power to emergency buses is supplied by either one of two EDGs. Power through one EDG meets the design success criteria of Chapter 15 of the Harris UFSAR for all accidents including LOCA/LOOP. In addition, the non-safety DSDG, a supplemental power source, can provide the required power to maintain the plant in hot shutdown.

The most limiting accident condition assumed in Chapter 15 of the Harris UFSAR is the LOCA/LOOP, where the non-safety DSDG cannot be credited. The design success criteria for this case is the power through one of two EDG feeding the emergency bus with all associated equipment of the emergency bus operating. For this case, the design success criteria are satisfied and adequate defense-in-depth is maintained for the worst-case scenario.

The NRC staff finds that the design success criteria will be met during the entry of the RICT Program for TS 3.8.1.1 Action d1. The staff examined the RMA examples related to TS 3.8.1.1 Action d and found that these RMAs are reasonable and consistent with the intent of NEI 06-09 Section 3.4.3 for risk awareness and control during the emergent scenarios for TS 3.8.1.1 Action d. Considering that the design success criteria will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.1.1 Action d1, the NRC staff finds that the proposed changes to TS 3.8.1.1 Action d1 is acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the AC sources to perform their safety functions (assuming no additional failures of electrical components) is maintained.

<u>LCO 3.8.1.1 A.C. Sources Operating - ACTION g - With Contiguous events of an offsite or an onsite A.C, source becoming inoperable and resulting in failure to meet the LCO</u>

Current Required Action g1

> Within 6 days, restore all A.C. sources required by 3.8.1.1 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

The licensee proposed the deletion of this TS Action and its required Action from the Harris TS to reflect a change proposed in the LAR dated July 25, 2019.[14] The proposed change was to eliminate a second Completion Time in TS 3/4.8.1.1, which is aligned with the justification provided in NRC-approved TSTF-439, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO," Revision 2, dated June 20, 2005.[15] The licensee stated these changes are consistent with NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 4, dated April 2012.[16] The NRC staff notes that this change is requested under a separate LAR and was reviewed separately by the NRC staff. The evaluation of this request is therefore not a part of this SE. The deletion of TS 3.8.1.1, Action g1 was approved by the NRC staff in a letter dated June 29, 2020, issuing Amendment No. 177[17] in response to the earlier LAR.

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[14] ADAMS Accession No. ML19206A599
[15] ADAMS Accession No. ML051860296
[16] ADAMS Accession No. ML12100A222
[17] ADAMS Accession No. ML20099F505

<u>LCO 3.8.1.1 A.C. Sources Operating - ACTION h - With one automatic load sequencer inoperable</u>

Current Required Action h1

> Restore the automatic load sequencer to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Proposed Required Action h1

> Restore the automatic load sequencer to OPERABLE status within 24 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

The licensee has requested to use the RICT Program for TS 3.8.1.1 Action h1. For TS 3.8.1.1 Action h1, per the LAR, an estimated RICT of 30 days is calculated. The licensee stated in its supplement dated July 27, 2020, that the approximate range determined for RICT depending on the possible emergent conditions is estimated from 2.1 days to 30 days.

As described in Section 7.3.1.5 of the UFSAR, there are separate but identical sequencers for each safety train (A and B). All components of each sequencer (exclusive of inputs from external sensing devices, Main Control Board displays and controls and transfer switches) are in a single cabinet. The train A sequencer is powered from 125 V DC Distribution Panel 1A-SA and the Train B sequencer is powered from 125 V DC Distribution Panel 1B-SB. Section 8.3.1.1.2.8 of the UFSAR describes the automatic tripping and loading of ESF buses. The EDGs and the emergency load sequencers will be started if an ESF actuation signal is present. The loads are connected sequentially through the load sequencer to minimize the effect of an excessive voltage drop on the safety buses due to starting large motors when no offsite power is available and all other plant loads remain connected. If offsite power is available, the appropriate safety-related loads will be started sequentially on this preferred offsite power source.

The NRC staff notes that with one automatic load sequencer inoperable, the remaining load sequencer will be available to perform the required function. As described in Section 8.3.1.1.2.4 of the UFSAR, for LOOP, when the safety injection signal is not present, the required safety and non-safety loads blocks can be manually switched to emergency buses. Therefore, for all LOOP events not involving LOCA, the manual actions are credited for connecting and tripping the required loads. In addition, the DSDG, a supplemental power source, can provide the required power to maintain the plant in hot shutdown during a LOOP event with ESF actuation not present.

The most limiting accident condition is LOOP when safety injection signal is present. For this condition, the EDG would auto-start and the non-safety station service transformer fed from the ESF bus would be automatically tripped (due to a safety injection signal). Concurrently the sequencer will shift to the LOCA with LOOP program. The EDG breaker will close upon sensing rated frequency and voltage at the EDG and after an elapse of a time delay for residual voltage to decay. The ESF loads will then be sequenced onto the diesel generator by the remaining sequencer. Even though there is only a single layer of defense-in-depth for the TS 3.8.1.1 ACTION h for the worst case scenario (i.e., LOCA coincident with LOOP), the NRC staff finds that this is reasonable and acceptable because of the low probability of the worst case accident, and the implementation of RMAs during the RICT Program entry for TS 3.8.1.1 Action h1.

The NRC staff finds that the design success criteria will be met during the entry of the RICT Program for TS 3.8.1.1 Action h1. The staff examined the RMA examples related to TS 3.8.1.1 Action h and found that these RMAs are reasonable and consistent with the intent of NEI 06-09 Section 3.4.3 for risk awareness and control during the emergent scenarios for TS 3.8.1.1 Action h. Considering that the design success criteria will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.1.1 Action h1, the NRC staff finds that the proposed changes to TS 3.8.1.1 Action h1 is acceptable because the plant will continue to have minimum defense-in-depth in which the capability of the AC sources and supported equipment to perform their safety functions (assuming no additional failures of electrical components) is maintained.

<u>LCO 3.8.2.1 D.C. Sources Operating</u>

Current Required Action

> With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Proposed Required Action

> With one of the required D.C. electrical sources inoperable, restore the inoperable D.C. electrical source to OPERABLE status within 2 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

This TS Action corresponds to TSTF-505, Condition 3.8.4, Required Action B.1. This LCO only relates to loss of DC due to loss of battery supply in one train of DC. Although Modes 1, 2, 3 and 4 are specified for this Action, the RICTs are only applied for modes 1 and 2 as stated in the LAR. The licensee has requested to use the RICT Program for the TS 3.8.2.1 Action. The licensee stated in its supplement dated July 27, 2020, that the approximate range determined for RICT depending on possible emergent conditions is estimated from 7.2 days to 30 days for the DC source for division SA.

There are two DC divisions (SA and SB) at Harris. Harris has two independent DC buses (1A-SA and 1B-SB), each with two dedicated battery chargers and a dedicated battery set. The battery chargers are fed through two emergency 480V MCCs. Each DC division has a separate battery and battery charging system. There is no connection (or cross-tie) between the battery and charger set of one division and their redundant counterparts.

The DC power is supplied through battery chargers when AC is available. Each battery train supplies the required DC power for starting the associated EDG. The loads associated with EDGs A and B are fed from the DC divisions SA and SB, respectively. Two PORVs, 1RC-119 and 1RC-114, are also fed from DC divisions SA and SB, respectively. The loads associated with turbine driven auxiliary feed water pump, however, are divided between both DC divisions SA and SB. During loss of DC division SB (i.e., DP-1B-SB), the turbine driven auxiliary feed water pump cannot be operated from the main control room; it will require local operation.

Therefore, the loss of DC division SB is expected to have slightly larger impact and consequently a shorter RICT estimate.

As described in Section 8.3.2.1 and Figure 8.1.3-3 of the UFSAR, the two banks of 125V batteries (designated 1A-SA and 1B-SB), their associated load centers and distribution panels have been arranged to feed the safety-related DC loads associated with divisions SA and SB respectively. The two independent DC buses (1A-SA and 1B-SB) each has two dedicated battery chargers and a dedicated battery bank. The battery chargers are fed through two emergency 480V MCCs. During this LCO, the power to the affected DC bus is provided via one of the two battery chargers when the associated 480V ESF AC bus is energized. At the onset of a LOOP event, this DC bus is considered de-energized since AC power will be lost to its chargers. Furthermore, the associated EDG supported by this DC bus cannot operate. With the associated EDG not operating, one division of AC and DC will be lost during LOOP. Similar conditions are also anticipated during the LOOP/LOCA initiator. During LOCA scenarios with offsite power available, the affected DC bus will remain energized fed by one or more battery chargers. The licensee stated in the RAI response dated November 11, 2020, that the battery chargers do not have enough capacity to support LOOP, LOCA and LOOP/LOCA loads. The NRC staff notes that for other events except LOOP, LOCA and LOOP/LOCA, both DC buses can be considered available: one fed through either battery or chargers and the other fed through chargers.

The most limiting accident condition assumed in Chapter 15 of Harris UFSAR is the LOOP/LOCA, where only one remaining DC division to support the required loads would be available. For this case, redundancy is reduced to one remaining DC division. Although there is only a single layer of defense-in-depth for the TS 3.8.2.1 Action worst scenario, by considering the implementation of RMAs during the RICT Program entry for TS 3.8.2.1 Action, the NRC staff finds that this is reasonable and acceptable.

The NRC staff finds that the design success criterion will be met during the entry of the RICT Program for the TS 3.8.2.1 Action. The staff examined the RMA examples related to TS 3.8.2.1 Action and found that these RMAs are reasonable and consistent with the intent of NEI 06-09, Section 3.4.3 for risk awareness and control during the emergent scenarios for the TS 3.8.2.1 Action.

Considering that the design success criterion will be met and RMAs will be implemented during the RICT Program entry for the TS 3.8.2.1 Action, the NRC staff finds that the proposed changes to TS 3.8.2.1 Action are acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the DC sources to perform their safety functions (assuming no additional failures of electrical components) is maintained.

LCO 3.8.3.1 Onsite Power Distribution Operating - ACTION a - With one of the required divisions of A.C. ESF buses not fully energized

Current Required Action a

> Reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Proposed Required Action a

> Reenergize the division within 8 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The licensee has requested to use the RICT Program for TS 3.8.3.1 Action a. For this TS Action, per the LAR, an estimated RICT of 30 days is calculated. The licensee stated in its supplement dated July 27, 2020, that the approximate range determined for RICT depending on possible emergent conditions is estimated from 10.7 days to 30 days for each AC ESF Bus. The onsite AC system includes two 6.9 kV ESF buses (1A-SA and 1B-SB), two EDGs, four 480 V buses (two of the four are fed from each ESF bus 1A-SA and 1B-SB). The 480 V buses supply loads directly and through several MCCs. The NRC staff notes that the electrical distribution and the load listing for the two AC ESF electrical divisions are independent and symmetric even considering the plant safety systems composed of three trains. With one of the required AC ESF division fully energized and available, the design success criteria will be met for all required functions during LOOP and LOOP/LOCA scenarios.

The NRC staff also notes that with one division of AC ESF buses not energized, the remaining division of AC ESF buses will be available to meet the ESF loads. The condition constitutes a reduced redundancy level of ESF AC bus divisions to one division. Even though there could be only a single layer of defense-in-depth for the TS 3.8.3.1 Action a, by considering the implementation of RMAs during the RICT Program entry for TS 3.8.3.1 Action a, the NRC staff finds that this is reasonable and acceptable.

The NRC staff finds that the design success criteria will be met during the entry of the RICT Program for TS 3.8.3.1 Action a. The NRC staff examined the RMA examples related to TS 3.8.3.1 Action a and found that these RMAs are reasonable and consistent with the intent of NEI 06-09 Section 3.4.3 for risk awareness and control during the emergent scenarios for TS 3.8.2.1 Action a.

Considering that the design success criteria will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.3.1 Action a, the NRC staff finds that the proposed changes to TS 3.8.3.1 Action a is acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the power distribution system to perform their safety functions (assuming no additional failures of electrical components) is maintained.

<u>LCO 3.8.3.1 Onsite Power Distribution Operating - ACTION b - With one 118-volt A.C. vital bus not energized from its associated inverter</u>

Current Required Action b

> Reenergize the 118-volt A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Proposed Required Action b

> Reenergize the 118-volt A.C. vital bus within 2 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The licensee has requested to use the RICT Program for TS 3.8.3.1 Action b. However, by the licensee's current calculations, the use of the RICT Program for this action is precluded by the instantaneous CDF or LERF limits of 1E-03 or 1E-04, respectively. The licensee has requested that this TS Action remain within the scope of the LAR, and it is proposed that the RICT Program be used on this TS Action if the plant risk estimates decrease in the future for TS 3.8.3.1 Action b.

As described in Section 8.3.1.1.1.4 of the UFSAR, the 118V ESF uninterruptible AC system consists of four separate rectifier/inverters channels. Channels I and III feed vital bus 1A-SA and channels II and IV feed vital bus 1B-SB. The major loads of 118 VAC channels include safety-related Reactor Protection System (RPS), Nuclear Instrumentation System, the Post-accident Monitoring System and the other safety-related loads associated with vital instrumentation and controls. The loads are distributed between four channels in a manner such that with one of two vital AC buses not energized, the remaining redundant and independent bus can accommodate all the required loads to meet the design success criterion for all types of accidents. Even though there is only a single layer of defense-in-depth for the TS 3.8.3.1 Action b, by considering the implementation of RMAs during the RICT Program entry for TS 3.8.3.1 Action b, the NRC staff finds that this is reasonable and acceptable.
The NRC staff notes that with one out of two vital AC buses not energized, the remaining bus fed from two independent channels of rectifier/inverters will be available to accommodate the required loads for all types of accidents including LOCAs. The condition constitutes a reduced redundancy level for remaining one vital AC bus.

The NRC staff finds that the design success criteria will be met during the entry of the RICT Program for TS 3.8.1.1 Action b. The staff concluded that the licensee may implement the RICT Program on this TS Action if the instantaneous CDF or LERF limits of 1E-03 or 1E-04, respectively are met.

Considering that the design success criteria will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.3.1 Action b, the NRC staff finds that, if the instantaneous CDF or LERF limits of 1E-03 or 1E-04 respectively are met, the proposed changes to TS 3.8.3.1 Action b are acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the power distribution system to perform its safety functions (assuming no additional failures of electrical components) is maintained.

<u>LCO 3.8.3.1 Onsite Power Distribution Operating - Action c - With one 118-volt A.C. vital bus not energized from its associated inverter</u>

Current Required Action c

> Re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

Proposed Required Action c

> Re-energize the 118-volt A.C. vital bus through its associated inverter connected to its associated D.C. bus within 24 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

The licensee has requested to use the RICT Program for TS 3.8.3.1 Action c. For this TS Action, per the LAR, an estimated RICT of 30 days is calculated. The licensee stated in its supplement dated July 27, 2020, that the approximate range determined for the RICT depending on the possible emergent conditions is estimated from 10.7 days to 30 days for each 118 V vital AC bus.

As described in Section 8.3.1.1.1.4 of the UFSAR, the 118V ESF uninterruptible AC system consists of four separate rectifier/inverters channels. Channels I and III feed vital bus 1A-SA and channels II and IV feed vital bus 1B-SB. Each safety 7.5 KVA inverter is nominally rated 118 VAC, 60 Hz, and is normally supplied through its rectifier from a 480V ESF MCC. Should the voltage drop below the required level, the power for the inverter is supplied automatically from a 125 VDC bus. Channels I and III are powered from MCC 1A21-SA and MCC 1A31-SA respectively for the main feeds and from the 125 VDC bus DP- 1A-SA for backup. Channels II and IV are powered from MCC 1B21-SB and MCC 1B31-SB respectively for the main feeds and from 125V DC bus DP-1B-SB for backup. During this LCO, both vital AC bus will be available; one bus would be fed through both the DC bus and AC bus, and the other bus would be supplied through its rectifiers from the 480V ESF MCCs. The LCO bus is considered lost when the power to the associated 480V ESF bus is lost.

For the LOOP and LOOP/LOCA initiators, the LCO bus is de-energized. The remaining redundant and independent 118 ESF uninterruptible AC bus, however, can support one train of all vital AC loads for all initiators considered in Chapter 15 of the Harris UFSAR. For this case, the redundancy is reduced to one remaining DC division. Although there is only a single layer of defense-in-depth for the TS 3.8.3.1 Action c for the worst-case scenario, by considering the implementation of RMAs during the RICT Program entry for TS 3.8.3.1 Action c, the NRC staff finds that this is reasonable and acceptable.

The NRC staff notes that with one out of two vital AC buses not energized through its associated inverter connected to its associated DC bus, the remaining redundant bus will be available for all accidents including LOOP/LOCAs to meet design success criteria requirements. This condition constitutes a reduced redundancy level for remaining one vital AC bus. Even though there is only a single layer of defense-in-depth for the TS 3.8.3.1 Action c, by considering the implementation of RMAs during the RICT Program entry for TS 3.8.3.1 Action c, the NRC staff finds that this is reasonable and acceptable.

The NRC staff finds that the design success criteria will be met during the entry of the RICT Program for TS 3.8.3.1 Action c. The staff examined the RMA examples related to TS 3.8.3.1 Action c and found that these RMAs are reasonable and consistent with the intent of NEI 06-09 Section 3.4.3 for risk awareness and control during the emergent scenarios for TS 3.8.2.1 Action c. Considering that the design success criteria will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.3.1 Action c, the NRC staff finds that the proposed changes to TS 3.8.3.1 Action c are acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the power distribution system to perform their safety functions (assuming no additional failures of electrical components) is maintained.

LCO 3.8.3.1 Onsite Power Distribution Operating - Action d - With either 125-volt D.C. bus 1A-SA or 1B-SB not energized from its associated Emergency Battery

Current Required Action d

> Reenergize the D.C. bus from its associated Emergency Battery within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Proposed Required Action d

> Reenergize the D.C. bus from its associated Emergency Battery within 2 hours *or in accordance with Risk-Informed Completion Time Program* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

The licensee has proposed the use of RICT Program for this TS action, but by the licensee's current calculation, the use of the RICT Program on this TS action is precluded by the instantaneous CDF or LERF limits of 1E-03 or 1E-04, respectively. The licensee has requested that this TS action remain within the scope of the LAR, and it is proposed that the RICT Program be used for this TS action if plant risk estimates decrease in the future for TS 3.8.3.1 Action d.

As described in Section 8.3.2.1 and Figure 8.1.3-3 of the UFSAR, the two banks of 125V batteries (designated 1A-SA and 1B-SB), their associated load centers and distribution panels have been arranged to feed the safety-related DC loads associated with divisions A and B respectively. The two independent DC buses (1A-SA and 1B-SB) each has two dedicated battery chargers and a dedicated battery bank. The battery chargers are fed through two emergency 480V MCC.

During this LCO, the power to the affected DC bus (i.e., the bus not energized by its battery) is provided via one out of the two battery chargers when the associated 480V ESF AC bus is energized. At the onset of a LOOP event, this DC bus is considered de-energized since AC power will be lost to its chargers. Furthermore, the associated EDG supported by this DC bus cannot operate. With the associated EDG not operating, one division of AC and DC will be lost during LOOP. Similar conditions are also anticipated during LOOP/LOCA initiator.

In its response to RAI 7 in its supplement dated November 11, 2020, the licensee stated that the battery chargers do not have enough capacity to support LOCA loads. The NRC staff notes that for all other events except LOOP, LOCA and LOOP/LOCA, both DC buses can be considered available, one fed through either its dedicated battery or chargers and the other fed through its chargers. The NRC staff notes that for these accident conditions (excluding LOOP, LOOP/LOCA, and LOCAs), the design success criteria will be met by either of the two redundant DC trains, or by the additional power from the non-safety DSDG.

The NRC staff notes that for the most limiting accident conditions; LOOP, LOCA and LOOP/LOCA, there will be one remaining DC bus energized and available to meet design success criteria requirements. This condition constitutes a reduced redundancy level for DC bus divisions to one division. Even though there is only a single layer of defense-in-depth for the TS 3.8.3.1 Action d, by considering the implementation of RMAs during the RICT Program entry for TS 3.8.3.1 Action d, the NRC staff finds that this is reasonable and acceptable.

The NRC staff finds that the design success criteria will be met during the entry of the RICT Program for TS 3.8.1.1 Action d. The staff concluded that the licensee may implement RICT Program on this TS Action if the instantaneous CDF or LERF limits of 1E-03 or 1E-04, respectively are met. Considering that the design success criteria will be met and RMAs will be implemented during the RICT Program entry for TS 3.8.3.1 Action d, the NRC staff finds that, if the instantaneous CDF or LERF limits of 1E-03 or 1E-04 respectively are met, the proposed changes to TS 3.8.3.1 Action d are acceptable because the plant will continue to have adequate defense-in-depth, in which the capability of the power distribution system to perform its safety functions (assuming no additional failures of electrical components) is maintained.

3.1.2.2.3 Electrical Conclusion

The NRC staff finds that while the redundancy is not maintained (e.g., one train of a two-train system is inoperable), the CT extensions in accordance with the RICT Program are acceptable because (a) the capability of the systems to perform their safety functions (assuming no additional failures) is maintained, and (b) the licensee has satisfactorily demonstrated that identifying and implementing compensatory measures or RMAs, in accordance with the RICT Program, are appropriate to monitor and control risk.

3.1.2.3 Evaluation of Instrumentation and Control Systems

The licensee requested to use the RICT Program to extend the existing CT for the following TS conditions. The NRC staff's evaluation of the proposed changes considered a number of potential plant conditions allowed by the new TSs and considered what redundant or diverse

means were available to assist the licensee in responding to various plant conditions. The plant conditions evaluated are discussed in more detail below.

The NRC staff followed the guidance of RG 1.174 and RG 1.177 to assess the proposed changes consistent with the defense-in-depth criteria. The applicable defense-in-depth criteria that affect Harris Instrumentation and Control systems are:

- Over-reliance on programmatic activities as compensatory measures associated with the change in the licensing basis is avoided.

- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system (e.g., there are no risk outliers).

- Defenses against potential CCF are maintained and the potential for the introduction of new CCF mechanisms is assessed.

- The intent of the plant's design criteria is maintained.

Based on its review, the NRC staff verified that, in accordance with the Harris UFSAR, in all applicable operating modes, the affected protective feature would perform its intended function by ensuring the ability to detect and mitigate the associated event or accident when the CT of a channel is extended. However, it is recognized that while in an ACTION statement, redundancy of the given protective feature will be temporarily reduced, and, accordingly, the system reliability will be reduced. The NRC staff examined the design information from the Harris

UFSAR and the risk-informed LCO Conditions for the affected Instrumentation and Control functions. Based on this information, NRC staff confirmed that under any given DBA evaluated in the Harris UFSAR, the affected Instrumentation and Control protective features maintain adequate defense-in-depth by either necessary redundancy (e.g. at least one redundant channels) and/or necessary diversity (e.g. at least one alternative safety features).

Since the licensee did not propose any changes to the design basis, the independency and the fail-safe principle remain unchanged. The NRC staff reviewed the licensee's proposed TS changes to assess the availability of the redundant or diverse means to accomplish the safety functions. The NRC concludes that the proposed changes do not alter the ways in which the Harris Instrumentation and Control systems fail, do not introduce new CCF modes, and system independence is maintained. The NRC staff finds that while some proposed changes reduce the level of redundancy of the affected Instrumentation and Control systems during the proposed RICT period, this reduction may reduce the level of defense against some CCFs; however, the NRC staff finds, as described below, that such reduction in redundancy and defense against CCFs during the RICT period are acceptable due to existing diverse means available to maintain adequate defense-in-depth against a potential single failure during the RICT for the Harris Instrumentation and Control systems.

The following summarizes this section of the NRC staff's determinations with respect to the defense-in-depth principle for the functions identified in the LAR by identifying associated diverse means that maintain adequate defense-in-depth against potential single failure during the RICT for the Harris Instrumentation and Control systems. The change to the TS that reflects applying RICT to the associated ACTIONs is indicated in italics.

3.1.2.3.1 <u>LCO 3.3.1 Reactor Trip System Instrumentation</u>

LCO 3.3.1 requires that "[a]s a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE." As described in Harris UFSAR Section 7.2 "Reactor Trip System" the RPS contains two redundant logic trains (A and B), which receive inputs from the analog protection channels to complete the logic necessary to automatically open the reactor trip breakers. The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset level.

<u>FUNCTIONAL UNIT 1. Manual Reactor Trip</u>

The proposed RICT Program modifies ACTION 1 as "[w]ith the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement. restore the inoperable channel to OPERABLE status within 48 hours *or in accordance with the Risk-Informed Completion Time Program*, or be in HOT STANDBY within the next 6 hours." TABLE 3.3-1, "Reactor Trip System Instrumentation," of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to manually trip the reactor. ACTION 1 is applied to this functional unit. Under modified ACTION 1 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the RPS trip capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 1 for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means available. The NRC staff concludes that this proposed change maintains the RPS trip capabilities and is consistent with the defense-in-depth principle.

<u>FUNCTIONAL UNIT 2.a. Power Range, Neutron Flux High Setpoint</u>
<u>FUNCTIONAL UNIT 2.b. Power Range, Neutron Flux Low Setpoint</u>
<u>FUNCTIONAL UNIT 3. Power Range, Neutron Flux High Positive Rate</u>
<u>FUNCTIONAL UNIT 4. Power Range, Neutron Flux, High Negative Rate</u>

The proposed RICT Program modifies ACTION 2 as "[w]ith the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 6 hours, *or in accordance with the Risk-Informed Completion Time Program."* In TABLE 3.3-1 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, these functional units are identical in Total Number of Channels, CHANNELS TO TRIP, and Minimum Channels OPERABLE. The loss of function and defense-in-depth analyses to the proposed changes are identical and are illustrated using FUNCTIONAL UNIT 2.a. In TABLE 3.3-1 of Attachment 2 of the LAR, the Total Number of Channels is 4, and the CHANNELS TO TRIP is 2.

In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of four. ACTION 2 is applied to these functional units. Under modified ACTION 2.a., the number of operable channels is 3 (one less than the Total Number of Channels). The three operable channels maintain the RPS trip capability.

In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 2.a., FUNCTIONAL UNIT 2.b., FUNCTIONAL UNIT 3., and FUNCTIONAL UNIT 4., for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 2.a, FUNCTIONAL UNIT 2.b., FUNCTIONAL UNIT 3., and FUNCTIONAL UNIT 4 maintain the RPS trip capabilities and are consistent with the defense-in-depth principle.

<u>FUNCTIONAL UNIT 7. Overtemperature ΔT</u>
<u>FUNCTIONAL UNIT 8. Overpower ΔT</u>
<u>FUNCTIONAL UNIT 9. Pressurizer Pressure--Low (Above P-7)</u>
<u>FUNCTIONAL UNIT 10. Pressurizer Pressure—High</u>
<u>FUNCTIONAL UNIT 11. Pressurizer Water Level —High (Above P-7)</u>

The proposed RICT Program modifies ACTION 6 as "[w]ith the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 6 hours, *or in accordance with the Risk-Informed Completion Time Program."* In TABLE 3.3-1 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, these functional units are identical in Total Number of Channels, CHANNELS TO TRIP, Minimum Channels OPERABLE. The loss of function analyses to the proposed changes are identical and are illustrated using the FUNCTIONAL UNIT 7.

In TABLE 3.3-1 of Attachment 2 of the LAR, the Total Number of Channels is 3, and the CHANNELS TO TRIP is 2. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three. ACTION 6 is applied to these functional units. Under modified ACTION 6, the number of operable channels is 2 (one less than the Total Number of Channels). The two operable channels maintain the RPS trip capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 7, FUNCTIONAL UNIT 8, FUNCTIONAL UNIT 9, FUNCTIONAL UNIT 10, and FUNCTIONAL UNIT 11, for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 7, FUNCTIONAL UNIT 8, FUNCTIONAL UNIT 9, FUNCTIONAL UNIT 10 and FUNCTIONAL UNIT 11 maintain the RPS trip capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 12.a. Reactor Coolant Flow -- Low Single Loop (Above P-8)

In TABLE 3.3-1 of Attachment 2 of the LAR, the Total Number of Channels is 3 per loop, and the CHANNELS TO TRIP is 2 channels in any operating loop. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three. ACTION 6 is applied to this functional unit. Under modified ACTION 6, the number of operable channels is 2 for the loop that under action (one less than the Total Number of Channels). The two operable channels maintain the RPS trip capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 12.a. for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 12.a maintains the RPS trip capabilities and is consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 13. Steam Generator Water Level – Low – Low

In TABLE 3.3-1 of Attachment 2 of the LAR, the Total Number of Channels is 3 per steam generator, and the CHANNELS TO TRIP is 2 channels in any operating steam generator. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three. ACTION 6 is applied to this functional unit. Under modified ACTION 6, the number of operable channels is 2 for the generator that under action (one less than the Total Number of Channels). The two operable channels maintain the RPS trip capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 13 for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 13 maintains the RPS trip capabilities and is consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 14. Steam Generator Water Level—Low Coincident With Steam/Feedwater Flow Mismatch

In accordance with the Harris UFSAR Section 7.2.1, Section 5 of Enclosure 1 of the LAR, and TABLE 3.3-1 of Attachment 2 of the LAR, for one steam generator, this functional unit consists of a 1 out of 2 sub logic for two steam generator level channels, and a 1 out of 2 sub logic for two steam/feedwater flow mismatch channels, and a 2 out of 2 logic that combines above two 1 out of 2 sub logics. ACTION 6 is applied to this functional unit. Under ACTION 6 only 1 channel is inoperable so both sub logics function and 2 out of 2 trip logic maintains its RPS trip capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 14 for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 14 maintain the RPS trip capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 15. Undervoltage--Reactor Coolant Pumps (Above P-7)

In accordance with the Harris UFSAR Section 7.2.1, Section 5 of Enclosure 1 of the LAR, and TABLE 3.3-1 of Attachment 2 of the LAR, this functional unit consists of 3 trip inputs from 3 RCPs with a 2 out of 3 logic. Each RCP trip input comes from two undervoltage relays with a 2 out of 2 logic. ACTION 6 is applied to this functional unit. Under modified ACTION 6 one undervoltage channel (relay) is inoperable which leads to one RCP trip input inoperable. The remaining 2 RCP trip inputs, however, maintain the RPS trip capability in the event of undervoltages at two out of three reactor coolant pump motor feeders. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 15 for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 15 maintain the RPS trip capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 16. Underfrequency - - Reactor Coolant Pumps (Above P-7)

In accordance with the Harris UFSAR Section 7.2.1, Section 5 of Enclosure 1 of the LAR, and TABLE 3.3-1 of Attachment 2 of the LAR, this functional unit consists of 3 trip inputs from 3 RCPs with a 2 out of 3 logic. Each RCP trip input comes from two underfrequency relays with a 2 out of 2 logic. ACTION 6 is applied to this functional unit. Under modified ACTION 6 one underfrequency channel (relay) is inoperable which leads to one RCP trip input inoperable. The remaining 2 RCP trip inputs, however, maintain the RPS trip capability in the event of underfrequency at two out of three reactor coolant pump motor feeders. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 16 for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 16 maintain the RPS trip capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 17.a. Turbine Trip (Above P-7) Low Fluid Oil Pressure

In TABLE 3.3-1 of Attachment 2 of the LAR, the Total Number of Channels is 3, and the CHANNELS TO TRIP is 2. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three. ACTION 6 is applied to this functional unit. Under modified ACTION 6, the number of operable channels is 2 (one less than the Total Number of Channels). The two operable channels maintain the RPS trip capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 17.a. for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 17.a maintain the RPS trip capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 20. Reactor Trip Breakers

The proposed RICT Program modifies ACTION 11 as "[w]ith one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours *or in accordance with the Risk-Informed Completion Time Program,* or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status." In TABLE 3.3-1 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, the RPS has two trains of reactor trip breakers (RTBs), and each train can initiate the reactor trip. Each train has two trip features of undervoltage and shunt trip attachment. Each feature can initiate the RPS trip. ACTION 11 is applied to this functional unit. Under modified ACTION 11, only one trip feature is inoperable, and the remaining trip feature maintains the RPS trip capability. In accordance with the Harris UFSAR Section 7.2.1, Section 5 of Enclosure 1 of the LAR, and TABLE 3.3-1 of Attachment 2 of the LAR, the reactor trip system (RTS) has two diverse trip features. For every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is one diverse mean available. The NRC staff concludes that the proposed change to FUNCTIONAL UNIT 20 maintains the RPS trip capabilities and is consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 21. Automatic Trip and Interlock Logic

The proposed RICT Program modifies ACTION 13 as "[w]ith the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program*, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1. provided the other channel is OPERABLE." In TABLE 3.3-1 of Attachment 2 of the LAR, the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to trip the reactor. ACTION 13 is applied to this functional unit. Under modified ACTION 13 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the RPS trip capability.

The licensee clarified in its supplement dated July 27, 2020, that the manual RTS actuation serves as the diverse means to the FUNCTIONAL UNIT 21, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 21 maintains the RPS trip capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

3.1.2.3.2 LCO 3.3.2 Engineered Safety Features Actuation System Instrumentation

LCO 3.3.2 requires that "[t]he Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4." As described in Harris UFSAR Section 7.3 "Engineered Safety Features System," the ESFAS consists of two distinct portions of circuitry: (1) An analog portion consisting of three to four redundant channels per parameter or variable which monitor various plant parameters; and (2) a digital portion consisting of two redundant logic trains which receive inputs from the analog protection channels and perform the logic needed to actuate ESF equipment. Each digital train is capable of actuating the required ESF equipment.

FUNCTIONAL UNIT 1. Safety Injection a. Manual Initiation

The proposed RICT Program modifies ACTION 18 as "[w]ith the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours *or in accordance with the Risk-Informed*

Completion Time Program, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours." TABLE 3.3-3, "Engineered Safety Features Actuation System Instrumentation," of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to manually initiate the safety injection. ACTION 18 is applied to this functional unit. Under modified ACTION 18 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the safety injection actuation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that other automatic ESFAS safety injection actuations are available as diverse means for FUNCTIONAL UNIT 1.a. The NRC staff concludes that this proposed change maintains the ESFAS safety injection actuation capabilities and is consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 1. Safety Injection b. Automatic Actuation Logic and Actuation Relays

The proposed RICT Program modifies ACTION 14 as "[w]ith the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program,* or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1. provided the other channel is OPERABLE."

In TABLE 3.3-3 of Attachment 2 of the LAR, the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to manually actuate the safety injection. ACTION 14 is applied to this functional unit. Under modified ACTION 14, the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the safety injection actuation capability.

The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS safety injection actuation serves as the diverse means to the FUNCTIONAL UNIT 1.b, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 1.b maintains the ESFAS safety injection actuation capabilities and is consistent with the defense-in-depth principle, without over-reliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 1. Safety Injection c. Containment Pressure--High-1
FUNCTIONAL UNIT 1. Safety Injection d. Pressurizer Pressure--Low
FUNCTIONAL UNIT 1. Safety Injection e. Steam Line Pressure--Low
FUNCTIONAL UNIT 4. Main Steam Line Isolation d. Steam Line Pressure – Low

The proposed RICT Program modifies ACTION 19 as "[w]ith the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours *or in accordance with the Risk-Informed Completion Time Program,"* and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1."

In TABLE 3.3-3 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, these functional units are identical in Total Number of Channels, CHANNELS TO TRIP, Minimum Channels OPERABLE and the design coincidence logic. The loss of function and defense-in-depth analyses to the proposed changes are identical and are illustrated using the FUNCTIONAL UNIT 1.c. In TABLE 3.3-3 of Attachment 2 of the LAR, the Total Number of Channels is 3, and the CHANNELS TO TRIP is 2. In accordance with the Section 5 of the LAR Enclosure 1, the trip coincidence of this functional unit is two out of three. ACTION 19 is applied to these functional units. Under modified ACTION 19, the number of operable channels is 2 (one less than the Total Number of Channels). The two operable channels maintain the safety injection initiation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 1.c., FUNCTIONAL UNIT 1.d., FUNCTIONAL UNIT 1.e., and FUNCTIONAL UNIT 4.d., for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 1.c, FUNCTIONAL UNIT 1.d, FUNCTIONAL UNIT 1.e., and FUNCTIONAL UNIT 4.d, maintain the ESFAS initiation capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 2. Containment Spray a. Manual Initiation
FUNCTIONAL UNIT 3. Containment Isolation b. Phase “B” Isolation 1) Manual Containment Spray Initiation

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1 (with 2 coincident switches). This means only one channel is required to manually actuate this safety function. ACTION 18 is applied to these functional units. Under modified ACTION 18 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Containment Spray initiation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified that other automatic ESFAS Containment Spray actuations are available as a diverse means for FUNCTIONAL UNIT 2.a. The NRC staff concludes that the proposed changes to FUNCTIONAL UNIT 2.a and FUNCTIONAL UNIT 3.b maintain the ESFAS Containment Spray actuation capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 2. Containment Spray b. Automatic Actuation Logic and Actuation Relays

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to actuate this safety function. ACTION 14 is applied to this functional units. Under modified ACTION 14 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Containment Spray actuation capability. The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS Containment Spray actuation serves as the diverse means to the FUNCTIONAL UNIT 2.b, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 2.b maintains the ESFAS initiation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 3. Containment Isolation a. Phase “A” Isolation 1) Manual Initiation

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1 (with 2 coincident switches). This means only one channel is required to manually initiate this safety function. ACTION 18 is applied to this functional units. Under modified ACTION 18 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Containment Isolation initiation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that other automatic ESFAS containment isolation actuations are available as diverse means for FUNCTIONAL UNIT 3.a.1). The NRC staff concludes that this proposed change maintains the ESFAS Containment Isolation initiation capabilities and is consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 3. Containment Isolation a. Phase “A” Isolation 2) Automatic Actuation Logic and Actuation Relays

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to initiate this safety function. ACTION 14 is applied to this functional units. Under modified ACTION 14 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Containment Isolation initiation capability. The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS Containment Isolation initiation serves as the diverse means to the FUNCTIONAL UNIT 3.a.2), and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 3.a.2) maintains the ESFAS initiation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 3. Containment Isolation b. Phase “B” Isolation 2) Automatic Actuation Logic and Actuation Relays

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to initiate this safety function. ACTION 14 is applied to this functional units. Under modified ACTION 14 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Containment Isolation initiation capability. The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS Containment Isolation actuation serves as the diverse means to the FUNCTIONAL UNIT 3.b.2), and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 3.b.2) maintains the ESFAS Containment Isolation initiation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 4. Main Steam Line Isolation a. Manual Initiation 2) System

The licensee proposed the new ACTION 27 as *“[w]ith the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or in accordance with the Risk-Informed Completion Time Program or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.”* TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Total Number of Channels is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to manually initiate this safety function. ACTION 27 is applied to this functional unit. Under ACTION 27 the number of operable channels is 1 (one less than the Total Number of Channels) during RICT. This one operable channel maintains the Main Steam Line Isolation actuation capability. In

Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that other automatic ESFAS Main Steam Line Isolation initiations are available as diverse means for FUNCTIONAL UNIT 4.a.2). The NRC staff concludes that this proposed change maintains the ESFAS Main Steam Line Isolation initiation capabilities and is consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 4. Main Steam Line Isolation b. Automatic Actuation Logic and Actuation Relays

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to initiate this safety function. ACTION 14 is applied to this functional unit. Under modified ACTION 14 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Main Steam Line Isolation initiation capability.

The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS actuation serves as the diverse means to the FUNCTIONAL UNIT 4.b, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 4.b maintains the ESFAS Main Steam Line Isolation actuation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 4. Main Steam Line Isolation c. Containment Pressure -- High - 2

In TABLE 3.3-3 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, the Total Number of Channels is 3, and the CHANNELS TO TRIP is 2. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three. ACTION 19 is applied to this functional unit. Under modified ACTION 19, the number of operable channels is 2 (one less than the Total Number of Channels). The two operable channels maintain the Main Steam Line Isolation initiation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 4.c. for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 4.c maintain the ESFAS Main Steam Line Isolation actuation capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 4. Main Steam Line Isolation e. Negative Steam Line Pressure Rate -- High

In TABLE 3.3-3 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, the Total Number of Channels is 3 per steam line, and the CHANNELS TO TRIP is 2 in any steam line. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three in any steam line. ACTION 19 is applied to this functional unit. Under modified ACTION 19, the number of operable channels is 2 in any steam line (one less than the Total Number of Channels). The two operable channels in any steam line maintain the Main Steam Line Isolation initiation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 4.e., for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 4.e maintain the ESFAS Main Steam Line Isolation initiation capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 5. Turbine Trip and Feedwater Isolation a. Automatic Actuation Logic and Actuation Relays

The proposed RICT Program modifies ACTION 24 as "[w]ith the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program,* or be in at least HOT STANDBY within the next 6 hours: however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE." TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to initiate this safety function. ACTION 24 is applied to this functional unit. Under modified ACTION 24 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the Turbine Trip and Feedwater Isolation initiation capability.

The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS Turbine Trip and Feedwater Isolation actuation serves as the diverse means to the FUNCTIONAL UNIT 5.a, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 5.a maintains the ESFAS Turbine Trip and Feedwater Isolation actuation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 5. Turbine Trip and Feedwater Isolation b. Steam Generator Water Level--High-High (P-14)

In TABLE 3.3-3 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, the Total Number of Channels is 4 per steam generator, and the CHANNELS TO TRIP is 2 in any steam generator. In accordance with the Section 5 of the LAR Enclosure 1, the trip coincidence of this functional unit is two out of four in any steam generator. ACTION 19 is applied to this functional unit. Under modified ACTION 19, the number of operable channels is 3 in any steam generator (one less than the Total Number of Channels). The three operable channels in any steam generator maintain the Line Turbine Trip and Feedwater Isolation actuation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 5.b. for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 5.b maintain the ESFAS initiation capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 6. Auxiliary Feedwater b. Automatic Actuation Logic and Actuation Relays

The proposed RICT Program modifies ACTION 21 as "[w]ith the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours *or in accordance with the Risk-Informed Completion Time Program,* or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours: however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE." TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to actuate the auxiliary feedwater (AFW). ACTION 21 is applied to this functional unit. Under modified ACTION 21 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the AFW actuation capability. The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS AFW actuation serves as the diverse means to the FUNCTIONAL UNIT 6.b, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 6.b maintains the ESFAS AFW actuation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

FUNCTIONAL UNIT 6. Auxiliary Feedwater c. Steam Generator Water Level – Low – Low 1) Start Motor-Driven Pumps
FUNCTIONAL UNIT 6. Auxiliary Feedwater c. Steam Generator Water Level – Low – Low 2) Start Turbine-Driven Pumps

In TABLE 3.3-3 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, these two functional units are identical in Total Number of Channels, CHANNELS TO TRIP, Minimum Channels OPERABLE. The loss of function and defense-in-depth analyses to the proposed changes are identical and are illustrated using the FUNCTIONAL UNIT 6.c.1). In TABLE 3.3-3 of Attachment 2 and Section 5 of Enclosure 1 of the LAR, the Total Number of Channels is 3 per steam generator for the functional unit of starting the motor-driven pumps, and the CHANNELS TO TRIP is 2 in any steam generator. In Section 5 of Enclosure 1 of the LAR, the trip coincidence of this functional unit is two out of three in any steam generator. ACTION 19 is applied to these functional unit. Under modified ACTION 19, the number of operable channels is 2 in any steam generator (one less than the Total Number of Channels). The two operable channels in any steam generator maintain the motor-driven pumps actuation capability. In Section 5 of Enclosure 1 of the LAR, the licensee confirmed, and the NRC staff verified, that for FUNCTIONAL UNIT 6.c.1) and FUNCTIONAL UNIT 6.c.2), for every Chapter 15 design basis accident that this affected FUNCTIONAL UNIT is credited for, there is at least one diverse means other than manual actuation available. The NRC staff concludes that these proposed changes to FUNCTIONAL UNIT 6.c.1) and FUNCTIONAL UNIT 6.c.2) maintain the ESFAS AFW actuation capabilities and are consistent with the defense-in-depth principle.

FUNCTIONAL UNIT 7. Safety Injection Switchover to Containment Sump a. Automatic Actuation Logic and Actuation Relays

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to initiate this safety function. ACTION 14 is applied to this functional unit. Under modified ACTION 14 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the actuation of this safety function. The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS actuation serves as the diverse means to the FUNCTIONAL UNIT 7.a, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 7.a maintains the ESFAS Safety Injection Switchover to Containment Sump actuation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

<u>FUNCTIONAL UNIT 8. Containment Spray Switchover to Containment Sump a. Automatic Actuation Logic and Actuation Relays</u>

TABLE 3.3-3 of Attachment 2 of the LAR, indicates that the Minimum Channels OPERABLE is 2, and the CHANNELS TO TRIP is 1. This means only one channel is required to initiate this safety function. ACTION 14 is applied to this functional unit. Under modified ACTION 14 the number of operable channels is 1 (one less than the Minimum Channels OPERABLE) during RICT. This one operable channel maintains the actuation of this safety function. The licensee clarified in its supplement dated July 27, 2020, that the manual ESFAS actuation serves as the diverse means to the FUNCTIONAL UNIT 8.a, and this manual action is modeled in the plant PRA, prescribed in the Harris Emergency Operating Procedures, or both. The NRC staff concludes that this proposed change to FUNCTIONAL UNIT 8.a maintains the ESFAS Containment Spray Switchover to Containment Sump actuation capabilities and is consistent with the defense-in-depth principle, without overreliance on programmatic activities as compensatory measures.

The NRC staff finds that the availability of the redundant or diverse protective features provide sufficient defense-in-depth to accomplish the safety functions, allowing for the extension of CTs in accordance with the RICT Program. The NRC staff finds that the licensee proposed RICT Program to the identified Instrumentation and Control systems is in compliance with 10 CFR 50.36(b) and 10 CFR 50.55a(h). The NRC staff reviewed the licensee's proposed TS changes and supporting documentation. The NRC staff finds that while the instrumentation and control redundancy is reduced, the CT extensions implemented in accordance with the RICT Program are acceptable because: (a) the capability of the instrumentation and control systems to perform their safety functions is maintained, (b) redundant or diverse means to accomplish the safety functions exist, and (c) the licensee will identify and implement RMAs to monitor and control risk in accordance with the RICT Program.

3.1.2.4 Key Principle 2 Conclusions

The LAR proposes to modify the TS requirements to permit extending selected CTs using the RICT Program in accordance with NEI 06-09-A. The NRC staff has reviewed the licensee's proposed TS changes and supporting documentation. The NRC staff finds that extending the selected CTs with the RICT Program following loss of redundancy, but maintaining the capability of the system to perform its safety function, is an acceptable reduction in defense-in-depth during the proposed RICT period provided that the licensee identifies and implements compensatory measures as appropriate during the extended CT.

Quantitative risk analysis, qualitative considerations including compensatory measures, and retaining the current CT for loss of all trains of a required system, assure that defense-in-depth is maintained to assure adequate protection of public health and safety. The NRC staff finds that the proposed changes are consistent with the defense-in-depth philosophy because:

- System redundancy (with the exceptions discussed above), independence, and diversity commensurate with the expected frequency and consequences of challenges to the system is preserved.

- Adequate capability of design features without an overreliance on programmatic activities as compensatory measures is preserved.

- The intent of the plant's design criteria continues to be met.

Therefore, NRC staff finds that these proposed changes meet the second key safety principle of RG 1.177 and are, therefore, acceptable. Additionally, the NRC staff concludes that the proposed changes are consistent with the defense-in-depth philosophy as described in RG 1.174.

3.1.3 Key Principle 3: Evaluation of Safety Margins

Section 2.2.2 of RG 1.177, Revision 1, states, in part, that sufficient safety margins are maintained when:

- Codes and standards … or alternatives approved for use by the NRC are met.
- Safety analysis acceptance criteria in the FSAR are met or proposed revisions provide sufficient margin to account for analysis and data uncertainties.

The licensee is not proposing in this application to change any quality standard, material, or operating specification. In the LAR, the licensee proposed to add a new program, "Risk-Informed Completion Time Program," in Section 6.8 of the TSs, which would require adherence to NEI 06-09-A.

The NRC staff evaluated the effect on safety margins when the RICT is applied to extend the CT up to a backstop of 30 days in a TS condition with sufficient trains remaining operable to fulfill the TS safety function. Although the licensee will be able to have design basis equipment out of service longer than the current TS allow, any increase in unavailability is expected to be insignificant and is addressed by the consideration of the single failure criterion in the design-basis analyses. Acceptance criteria for operability of equipment are not changed and, if sufficient trains remain operable to fulfill the TS safety function, the operability of the remaining train(s) ensures that the current safety margins are maintained. The NRC staff finds that if the specified TS safety function remains operable, sufficient safety margins would be maintained during the extended CT of the RICT Program. The NRC staff has evaluated specific proposed changes to the TS as described in Section 3.2 of this SE.

Safety margins are also maintained if PRA functionality is determined for the inoperable train which would result in an increased CT. Credit for PRA functionality, as described in NEI 06-09-A, is limited to the inoperable train, loop, or component. The reduced but available functionality may support a further increase in the CT consistent with the risk of the configuration. During this increased CT, the specified safety function is still being met by the operable train and therefore no evaluation of PRA functionality is required to meet the design basis success criteria.

3.1.3.1 Key Principle 3 Conclusions

The NRC staff finds that the design-basis analyses for Harris remain applicable. Although the licensee will be able to have design-basis equipment out-of-service longer than the current TS allow and the likelihood of successful fulfillment of the function will be decreased when redundant train(s) are not available, the capability to fulfill the function will be retained when the available equipment functions as designed. Any increase in unavailability because less equipment is available for a longer time is expected to be insignificant given the relatively short duration of the extended CT and the single failure criterion which is appropriately reflected in the RICT evaluation. Therefore, the staff finds that sufficient safety margins are maintained by the implementation of the RICT Program. The NRC staff concludes that the proposed changes meet the third key safety principle of RG 1.177 and are acceptable.

3.1.4 Key Principle 4: Change in Risk Consistent with the Safety Goal Policy Statement

Changes to the fixed TS CTs are typically evaluated by using the three-tiered approach described in Standard Review Plan (SRP) Chapter 16.1, "Risk-Informed Decision Making: Technical Specifications,"[18] RG 1.177, and RG 1.174, Revision 1. This approach addresses the calculated change in risk as measured by the change in ΔCDF and ΔLERF, as well as the ICCDP and ICLERP; the use of compensatory measures to reduce risk; and, the implementation of a CRMP to identify risk-significant plant configurations.

The NRC staff evaluated the licensee's processes and methodologies for determining that the change in risk from implementation of RICTs will be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The NRC staff evaluated the licensee's proposed changes against the three-tiered approach in RG 1.177, Revision 1, for the licensee's evaluation of the risk associated with a proposed TS CT change. The results of the staff's review are discussed below.

3.1.4.1 Tier 1: PRA Capability and Insights

Tier 1 evaluates the impact of the proposed changes on plant operational risk. The Tier 1 review involves two aspects: (1) the technical acceptability of the PRA models and their application to the proposed changes, and (2) a review of the PRA results and insights described in the licensee's application.

3.1.4.1.1 *PRA Acceptability*

The objective of the PRA acceptability review is to determine whether the Harris PRA that was used to implement the RICT Program is of sufficient scope, level of detail, and technical adequacy for this application.

The NRC staff evaluated the PRA acceptability information provided by the licensee in Enclosure 2 of its submittal, including industry peer review results and the licensee's self-assessment of the PRA models for internal events, internal flooding events, and fire events, against the guidance in RG 1.200, Revision 2. The licensee screened out all external hazard events, except for seismic, as described in Section 3.1.4.1.2 of this SE, as insignificant contributors to RICT calculations. The Harris PRA model with modifications is used as the CRMP model as described in Section 3.1.4.1.3 of this SE.

Internal Events and Internal Flooding PRAs

The NRC staff's review of the Harris internal events and internal flooding PRAs was based on the results of a full-scope peer review of the internal events PRA, self-assessments, focused-scope peer reviews, the associated facts and observations (F&Os) closure review described in Enclosure 2 of the LAR, and previously docketed information on PRA quality submitted to the NRC (1) to relocate surveillance frequencies to licensee control using TSTF-425, "Relocate Surveillance Frequencies to Licensee Control - RITSTF [Risk-Informed TSTF] Initiative 5b,"

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[18] ADAMS Accession No. ML070380228

Revision 3, approved November 29, 2016,[19] and (2) to adopt 10 CFR 50.69 risk-informed categorization, approved by letter dated September 17, 2019.

The March 2017 F&Os closure review for the Harris internal events PRA closed out all open findings, except for internal flooding related findings, by showing that the PRA met all applicable PRA Standard Supporting Requirements at Capability Category (CC) II. The NRC staff, in its SE for the LAR to adopt 10 CFR 50.69 risk-informed categorization, found that the March 2017 F&Os closure process for the Harris internal events PRA was performed consistent with guidance in the final revision of Appendix X,[20] dated February 21, 2017 to NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, dated November 2008,[21] and clarifications in the NRC staff's acceptance letter dated May 3, 2017.[22]

The March 2017 internal events PRA F&Os closure review did not result in the closure of all peer review and self-assessment findings associated with internal flooding. These open findings along with their dispositions for this application were provided by the licensee in Table E2-1 of Enclosure 2 of the LAR. As a result of the audit conducted by the NRC staff to ascertain the information needed to support its review of the application, the licensee further explained in the supplement dated July 27, 2020, that a focused-scope peer review was conducted in September 2019 for the high-level requirements IE, AS, SC, SY, QU, and LE of the PRA Standard, in accordance with the guidance in RG 1.200, Revision 2. A second F&Os closure review was performed in June 2020 on all finding-level F&Os from this peer review. The 2020 F&Os closure process was performed in accordance with the guidance in Appendix X of NEI 05-04 and clarifications in the NRC staff's acceptance letter dated May 3, 2017.

The licensee described in its supplement dated July 27, 2020, three changes to the internal events PRA and three changes to the internal flooding PRA that have been made since 2017 that are not associated with the resolutions of F&Os. The licensee assessed each of these changes to be PRA maintenance as defined in the PRA Standard, because the changes are data updates and/or utilize the same or similar methodologies currently implemented in the licensee's internal events PRA or internal flooding PRA, as applicable. In Enclosure 2 of the LAR, the licensee submitted ten finding-level F&Os related to the internal flooding PRA model that remained open, along with dispositions for this application, which the NRC staff reviewed. The licensee stated in its supplement dated July 27, 2020, that there are no open finding-level F&Os for the internal events PRA model.

The licensee's disposition for F&O 1-16 states, "[s]ince maintenance-induced flooding is not a significant contributor to CDF/LERF, and since Harris is a single unit site with no shared systems, it is expected that additional validation of the results will not impact CDF/LERF or the calculation of RICTs." In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 1 to provide information supporting the statement that maintenance-induced flooding is not a significant contributor to CDF and LERF. In its response to RAI 1a in its supplement dated November 11, 2020, the licensee explained that it conducted both a search of plant historical operating experience and discussions with operations staff, in order to determine if there are any maintenance-induced flooding events that should be included in the internal flooding PRA model. The licensee stated that, as a result of these activities, two maintenance-induced

[19] ADAMS Accession No. ML16200A285
[20] ADAMS Package Accession No. ML17086A431
[21] ADAMS Accession No. ML083430462
[22] ADAMS Accession No. ML17079A427

flooding events, that are applicable while the plant is operating, were identified and incorporated in the PRA model. As a result of the review conducted regarding the maintenance-induced flooding and the PRA model updated accordingly, the NRC staff concludes this F&O is sufficiently dispositioned for the RICT application.

The F&Os closure team did not close F&O 1-19 stating that the evaluation of alarms and alarm timing is not sufficient. The licensee performed a sensitivity study increasing this time by a factor of 3 and stated that there was "minimal impact on the flooding results." In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 1 to clarify if this assumption has been shown to not impact the RICT calculations and provide any supporting information. In its response to RAI 1b in its supplement dated November 11, 2020, the licensee explained that an operator interview was conducted to assess assumptions used in the human reliability analysis to credit flooding mitigation actions for Fire Protection and Normal Service Water/Emergency Service Water breaks. The licensee performed an assessment of these changes and determined that the human error probabilities either remain unchanged or decrease and that, therefore, they do not adversely impact RICT calculations. The NRC staff concludes this issue is sufficiently dispositioned for the RICT application.

Disposition for F&O 2-12 states, "F&O Closure panel recommended that a specific, combination-by-combination evaluation of the dependency should be provided to demonstrate that indeed there is insufficient dependency between these two groups of operator actions. This is a documentation issue only, and there is no impact on calculation of RICTs." In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 1 to provide detailed justification that there are no dependencies between the flood mitigation actions and the subsequent operator actions carried over from the internal events PRA. In its response to RAI 1c in its supplement dated November 11, 2020, the licensee stated it reperformed a dependency analysis of internal flooding and internal events operator actions credited in the PRA and identified one pair of actions of low dependency. The licensee indicated that the PRA was updated to use the calculated joint human error probability, if it was greater than the minimum joint human error probability. Since the combination-by-combination dependency analysis is incorporated into the PRA model, the NRC staff concludes this F&O is sufficiently dispositioned for the RICT application.

Fire Events PRA

The NRC staff review of the fire PRA was based on the results of a full-scope peer review of the fire PRA, a self-assessment, the associated F&Os closure review described in Enclosure 2 of the LAR, and previously docketed information on PRA quality submitted to the NRC (1) to relocate surveillance frequencies to licensee control using TSTF-425; (2) to adopt of National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," approved June 28, 2010,[23] and (3) to adopt 10 CFR 50.69 risk-informed categorization.

The October 2017 F&O closure review for the Harris fire PRA closed out many of the open findings by showing that the PRA met all applicable PRA Supporting Requirements at CC II. The NRC staff, in its SE for the LAR to adopt 10 CFR 50.69 risk-informed categorization, found that the October 2017 F&O closure process for the Harris fire PRA was performed consistent with guidance in Appendix X of NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer

[23] ADAMS Accession No. ML101750602

Review Process Guidelines," dated June 2010,[24] and clarifications in the NRC staff's acceptance letter dated May 3, 2017. The licensee stated that the Harris Fire PRA model was subject to a review conducted by the NRC staff during the NFPA 805 Pilot process and an additional focused-scope industry peer review, both in 2008 in accordance with ANSI/ANS-58.23-2007. Since the reviews of the Fire PRA model were performed prior to the publication of RG 1.200, Revision 2, a self-assessment was conducted to assess the differences between ANSI/ANS-58.23-2007 and the current version of the PRA Standard, ASME/ANS RA-Sa-2009. That assessment confirmed there were no technical differences between the two versions of the standard.

The licensee stated in its supplement dated July 27, 2020, that the Harris Fire PRA model was subject to a focused-scope peer review in June 2019 which addressed implementation of two upgrades to the model: incorporating credit for obstructed plume and resolution of finding FSS-F3. The finding-level F&Os generated from the June 2019 focused-scope peer review were reviewed and closed in June 2019 using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," dated August 2012,[25] as accepted by NRC in the letter dated May 3, 2017.

The open fire PRA findings along with their dispositions for this application were provided by the licensee in Table E2-1 of Enclosure 2 of the LAR. Table E2-2 of Enclosure 2 of the LAR presents findings that were assessed either as CC I or 'Not Met' but have no associated F&O. The NRC staff reviewed the dispositions to these findings along with additional information provided by the licensee about the resolution of these findings during the NRC staff's review of the licensee's LAR to adopt 10 CFR 50.69 risk-informed categorization. The NRC staff found that the dispositions for these open findings are acceptable for this application.

By letter dated September 17, 2019,[26] the NRC issued an amendment that added a new license condition to allow for the implementation of the provisions of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." As a result of the audit conducted by the NRC staff to ascertain the information needed to support its review of the application, the licensee stated in its supplement dated July 27, 2020, that the three fire PRA implementation items identified in the NRC staff's SE dated September 17, 2019 for the Harris LAR to adopt 10 CFR 50.69, have been implemented. The licensee also described three changes to the Fire PRA that have been made since 2017 that are not associated with the resolutions of F&Os. The licensee explained that one of the implementation items, related to the incorporation of credit for obstructed plumes per NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)," dated April 2016,[27] was assessed to be a PRA upgrade, which was subjected to the June 2019 focused-scope peer review.

PRA Technical Adequacy Conclusions

Based on the NRC staff's review of the licensee's submittal and assessments, the NRC staff concludes that the Harris PRA models for internal events, internal flooding events, and for fire events used to implement the RICT Program satisfy the guidance of RG 1.200, Revision 2. The NRC staff based this conclusion on the findings that the PRA models conform sufficiently to the

[24] ADAMS Accession No. ML102230070
[25] ADAMS Package Accession No. ML122400044
[26] ADAMS Accession No. ML19192A012
[27] ADAMS Accession Nos. ML16110A140 and ML16117A300

applicable industry PRA standards for internal events, internal flooding events, and for fire events at an appropriate capability category, considering the licensee's acceptable disposition of the peer review of F&Os and NRC staff review.

Based on the review of the provided information, the Harris PRA models were determined to be of sufficient technical adequacy to support implementation of the RICT Program. Therefore, the NRC staff finds that the licensee has satisfied the intent and regulatory guidance of Sections 2.3.1, 2.3.2, and 2.3.3 of RG 1.177, Revision 1, and Sections 2.3 and 2.5 of RG 1.174, Revision 3; and that the Harris PRA acceptability is sufficient to implement RMTS in accordance with the RICT Program and NEI 06-09-A.

PRA Update Process

Section 4.0 of the SE for the NEI 06-09-A requires the LAR to provide a discussion of the licensee's programs and procedures to ensure that the PRA models that provide the foundation for the RTR model are maintained consistent with the as-built, as-operated plant. The licensee has established a periodic update and review process for the PRAs that are used in the RTR model which is described in Enclosure 7 of the LAR. The NRC staff reviewed the licensee's PRA model update process to assess if the PRA models that support the RICT Program are maintained consistent with the as-built/as-operated and maintained plant.

The licensee indicated that its process is consistent with NEI 06-09-A. The licensee explained in Section 4.2 of Enclosure 7 of the LAR that its PRA update requirements include: (1) review of plant changes and discovered conditions for potential impact on the PRA models and the CRMP model including risk calculation to support the RICT Program, (2) review of plant changes that meet the plant procedure criteria for updating the PRA models, before the periodic update, (3) periodic update of the PRA models at least every two refueling cycle outages, and (4) performance of interim risk analyses or imposing administrative restrictions on use of the RICT Program, if significant plant changes or discovered conditions cannot be implemented immediately. The NRC staff noted that NEI 06-09-A specifies that "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations." It was also noted that Enclosure 7 of the LAR states that should a plant change or a discovered condition be identified that has a significant impact to the RICT Program calculations, as defined by plant procedure, an unscheduled update of the PRA model will be implemented. The licensee clarified in its supplement dated July 27, 2020, that changes are screened based on fleet procedures, which includes absolute or percentage increase in CDF/LERF, whichever is greater. The licensee clarified that a "significant impact" is defined in the plant procedure as a plant change that exceeds specified risk criteria that is consistent with industry practice.

The licensee stated in Section 4 of Enclosure 2 of the LAR that the update process includes review of plant changes, selected plant procedures, and plant operating data. The licensee stated that the extent of the model change will include consideration of accepted industry PRA practices, and assumptions consistent with those in the PRA Standard NRC guidance, advances in PRA technology and methodology, and changes in external hazard conditions. The NRC staff concludes the licensee's PRA model update process is acceptable because it is consistent with the regulatory guidance associated with RG 1.200 and NEI 06-09-A.

Risk Assessment Approaches and Methods

Changes to the PRA are expected to occur over time to reflect changes in PRA methods, and changes to the as-built, as-operated, and maintained plant to reflect the operating experience at the plant as specified in RG 1.200, Revision 2. Changes in PRA methods are addressed by constraint r. of TS Administrative Controls Section 6.8:

> The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the completion times must be PRA methods used to support this license amendment, or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

The NRC staff finds that this constraint is acceptable because it adequately implements the RICT Program using models, methods, and approaches consistent with applicable guidance that are acceptable to the NRC.

PRA Acceptability Conclusion

The licensee (1) has reviewed the PRA models using endorsed guidance and adequately addressed all identified issues for this application, (2) has established a periodic update and review process to update the PRA and associated CRMP model to incorporate changes made to the plant and PRA methods and data consistent with the RICT Program, and (3) will calculate RICTs using NRC-accepted PRA methods. Therefore, the NRC staff concludes that the licensee has and will maintain a PRA that is technically adequate to support implementation of the RICT Program.

3.1.4.1.2 *Scope of the PRA*

Topical Report NEI 06-09-A requires a quantitative assessment of the potential impact on risk due to impacts from internal and external events, including internal fires, internal floods, and significant external events. As discussed in Section 3.1.4.1.1 of this SE, the Harris PRA models used for the RICT Program include contributions from internal events, internal flooding events, and internal fire events. For external hazards for which a PRA is not available, the guidance in NEI 06-09-A allows for the use of bounding analysis of the risk contribution of the hazard for incorporation into the RICT calculation or justification for why the hazard is not significant to the RICT calculation. The licensee determined that all external hazards for which a PRA is not available are not significant to the risk calculation. The licensee provided its assessment of external hazard risk for the RICT Program in Enclosure 4 of the LAR. The licensee states that non-mandatory Appendix 6-A of the PRA Standard provides a guide for identification of most of the possible external events for a plant site. The NRC staff notes that this list is essentially the same list of hazards as presented in Table 4-1 of NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” Revision 1, dated March 2017.[28]

In Section 5 of Enclosure 4 of the LAR, the licensee stated that the overall process for addressing external hazards considers two aspects of the external hazard contribution to risk.

[28] ADAMS Accession No. ML17062A466

- The first aspect is the contribution from the occurrence of beyond design basis conditions, e.g., winds greater than design, seismic events greater than design basis earthquake, etc. These beyond design basis conditions challenge the capability of the SSCs to maintain functionality and support safe shutdown of the plant.

- The second aspect addressed are the challenges caused by external conditions that are within the design basis, but still require some plant response to assure safe shutdown, e.g., high winds or seismic events causing LOOP, etc. While the plant design basis assures that the safety related equipment necessary to respond to these challenges are protected, the occurrence of these conditions nevertheless cause a demand on these systems that presents a risk.

In Table E4-2 of Enclosure 4 of the LAR, the licensee provided a disposition for each non-seismic external hazard as well as other hazards and concludes that no unique PRA model for these hazards is required in order to assess configuration risk for the RICT Program (with the exception of internal flooding and internal fire, which are addressed by a PRA). The NRC staff notes that the preliminary screening criteria and progressive screening criteria presented in Table E4-3 of Enclosure 4 of the LAR, is the same criteria presented in supporting requirements for screening external hazards EXT-B1 and EXT-C1 of the PRA Standard.

External Hazards

The NRC staff's SE for NEI 06-09-A states that sources of risk besides internal events and internal fires (i.e., seismic and other external events) must be quantitatively assessed if they contribute significantly to configuration-specific risk. Bounding analyses or other conservative quantitative evaluations are permitted where realistic PRA models are unavailable. If sources of risk can be shown to be insignificant contributors to configuration risk, then they may be excluded from the RMTS.

In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 5 and RAI 6, about the actual values of these parameters used to calculate the seismic CDF of 2.14E-06/year in Section 6.1.4 of the Enclosure 4 to the LAR. Also, clarity was needed to ascertain whether the estimated seismic LERF based on a LERF-to-CDF ratios of 3.5% can be considered a bounding value. In its RAI response dated November 11, 2020, the licensee provided the basis for exclusion of external hazards from consideration in the determination of RICTs due to their insignificance to the calculation of configuration risk. The NRC staff reviewed Enclosure 4 to the LAR and the supplemental information to determine the acceptability of the consideration of risk from seismic events and other external hazards for this application. As discussed below, the NRC staff concluded that the licensee's justification for excluding external hazards from consideration in the RICTs was acceptable.

Seismic Hazard Contribution to the RICT

Following issuance of the NRC staff's letter dated March 12, 2012,[29] pursuant to 10 CFR 50.54(f), licensees were requested in a letter dated March 27, 2014[30] to reevaluate the seismic hazard at their sites against the current NRC requirements and guidance. The Harris site-specific hazard estimate was developed using Electric Power Research Institute (EPRI)

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[29] ADAMS Accession No. ML12053A340
[30] ADAMS Accession No. ML14090A441

Final Report 1025287, “Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” report dated February 2013. Using the updated seismic hazard for Harris, the licensee determined a bounding seismic CDF for Harris by applying the method that uses a plant level fragility presented in Appendix C of the NRC report on Generic Issue 199 dated September 2010, “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants.”[31] In reviewing that submittal, the NRC staff noted that the licensee estimated the seismic CDF without appropriately considering the seismic spectral ratios and estimated the seismic LERF without considering the site-specific seismic containment fragility.

In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 5 and RAI 6 about the licensee’s methodology for estimating the bounding seismic CDF and LERF, respectively. In its RAI 5 response dated November 11, 2020, the licensee explained that the plant level fragility that was used to estimate the seismic CDF is based on a high confidence of low probability of failure (HCLPF) of 0.29 g at the peak ground acceleration spectral frequency and a composite variability (β_c) of 0.4, from which a median capacity of 0.74 g was calculated. This plant level fragility is the same as that used in the NRC staff’s safety/risk assessment of Generic Issue 199, which is based on the Individual Plant Examination of External Events (IPEEE) for Harris. The licensee reviewed engineering changes for SSCs on the seismic safe shutdown equipment list made since 1990 and concluded that none of the changes impacted the functions of the SSCs. The licensee also determined that no SSCs had been added to the safe shutdown equipment list. Based on its review, the NRC staff verified that (1) the licensee’s assessment has no substantive changes to the safe shutdown equipment list and associated seismic capacities, and (2) the Harris IPEEE was performed and the plant level fragility is for the most limiting unscreened SSC assessed in the Harris IPEEE. Therefore, the NRC staff finds that the licensee’s plant level fragility is reasonable for estimating the bounding seismic CDF.

In its RAI 5 response dated November 11, 2020, the licensee indicated that it calculated the bounding seismic CDF using an approach similar to the “simple average” method described in the NRC safety/risk assessment report on Generic Issue 199. In this method, the seismic CDF was estimated for each of the five spectral frequencies (i.e., 1 Hz, 2.5 Hz, 5 Hz, 10 Hz, and peak ground acceleration) by convolving the reevaluated seismic hazard or Ground Motion Response Spectrum (GMRS) for each spectral frequency with the corresponding plant level fragility scaled to the respective spectral frequency. The fragility spectral ratios used in this calculation were those developed in the NRC safety/risk assessment report on Generic Issue 199, which was for the review level earthquake of 0.3g used in the Harris IPEEE. Based on the simple average of the calculated seismic CDF values for the five spectral frequencies, the licensee calculated a bounding seismic CDF of 4.98E-07 per year and an incremental core damage probability of 4.10E-08 for 30 days. Based on these results, the licensee concluded that the seismic CDF need not be included in the RICT calculations because it does not contribute significantly to the RICT calculations.

The NRC staff identified two issues with the licensee’s seismic CDF methodology. First, the use of five seismic CDF values in the averaging calculation is an increase over the four used in the NRC Generic Issue 199 report because the licensee included an estimated seismic CDF for 2.5 Hz. The licensee provided no justification for including this in the simple average calculation. Second, the licensee’s use of spectral ratios based on the review level earthquake is not appropriate because the spectral shape of the GMRS is different than that for the review level

[31] ADAMS Package Accession No. ML100270582

earthquake. Based on its review, the NRC staff determined that these issues with the licensee's methodology have a relatively small impact on the licensee's calculated seismic CDF. Therefore, the NRC staff concludes that the seismic CDF is an insignificant contributor to configuration risk and finds that not including seismic CDF in the RICT calculations is acceptable.

In its RAI 6 response dated November 11, 2020, the licensee provided the bounding seismic LERF using the "simple average" method as described above for seismic CDF. The licensee explained that the plant level fragility used to estimate the seismic LERF is based on a HCLPF of 0.57g at the peak ground acceleration spectral frequency, a composite variability of 0.35, and a median capacity of 1.2g. This plant level containment fragility was based on the results of a containment assessment that included (1) development of fragilities for SSCs critical to seismic LERF based on representative fragilities developed by EPRI for similar components at nuclear power plants, from which the most limiting fragility was used as the plant level fragility and (2) virtual walkdown of the key areas of the Harris Containment Building and Auxiliary Building of interest to the seismic LERF fragility assessment from which the licensee concluded that nothing was observed (e.g., seismic interactions, differential displacements, equipment or support degradations, etc.) that would negate the use of representative seismic capacities or fragilities assumed for the containment structure, valves, piping, and mechanical penetrations.

The licensee further explained in the response to RAI 6 that the seismic LERF was estimated for each of five different spectral frequencies (discussed above) by convolving the seismic CDF curves for each spectral frequency with the corresponding plant level seismic LERF fragility scaled to the respective spectral frequency. The fragility spectral ratios used in this calculation were developed by the licensee for each of the five spectral frequencies based on the GMRS for the Harris site. Based on the simple average of the calculated seismic LERF values for the five spectral frequencies, the licensee calculated a bounding seismic LERF of 9.97E-08 per year and an incremental large early release probability of 8.20E-09 for 30 days. Based on these results, the licensee concluded that the seismic LERF need not be included in the RICT calculations because it does not contribute significantly to the RICT calculations.

The NRC staff identified two issues with the licensee's seismic LERF methodology. First, as previously discussed for the seismic CDF, the use of five seismic LERF values in the averaging calculation is an increase over the four used in the NRC Generic Issue 199 report because the licensee included an estimated seismic LERF for 2.5 Hz. The licensee provided no justification for including this in the simple average calculation. Second, the licensee's use of representative fragilities is a significant source of uncertainty because (1) the RAI response does not address each of the factors that are included in acceptable methods, such as conservative deterministic failure margin, for developing plant-specific fragilities and (2) the video walkdown performed is not in accordance with Seismic Qualification Utilities Group procedures nor is it clear that the video walkdowns were performed by experienced seismic/structural engineers trained in performing walkdowns in accordance with Seismic Qualification Utilities Group procedures. Based on its review, the NRC staff determined, conservatively assuming the plant level seismic LERF fragility was the same as that used for the plant level seismic CDF fragility, that the seismic LERF is appreciably less than 5E-07 per year. Therefore, the NRC staff concludes that the bounding seismic LERF is an insignificant contributor to configuration risk and finds that not including seismic LERF in the RICT calculations is acceptable.

Extreme Winds and Tornado Hazards

Section 6 of Enclosure 4 to the LAR provides the licensee's evaluation of the impact on this application from the risk of extreme winds and tornadoes, which the licensee determined to be insignificant. The basis for the insignificant impact of extreme winds and tornadoes (including tornado-generated missiles) for this application relies on the design of SSCs and a tornado missile analysis. Table E4-2 of Enclosure 4 of the LAR indicates that Harris structures are designed to withstand design basis wind load and the effects of tornado missiles and meet the criteria in the 1975 SRP (which are included in the criteria presented in Appendix 6-A of the PRA Standard). Accordingly, the associated hazard screening criteria PS2 and C2 from Appendix 6-A of the PRA Standard were used to screen the extreme wind and tornado (including due to a hurricane) hazard.

In Section 6.2 of Enclosure 4 of the LAR, the licensee states that the site procedure for response to severe weather directs operations staff to place the plant in Mode 3 two hours prior to the anticipated arrival of sustained winds in excess of 74 miles per hour (mph) at the Harris site. Therefore, because the RICT Program cannot be implemented in Mode 3, hurricanes do not affect the RICT calculations and they were screened from inclusion in RICT calculations.

In Section 6.2.2 of Enclosure 4 of the LAR, the licensee states that the hazard from straight-line winds involve lower wind speeds that primarily result in a LOOP. Because LOOP events are already modeled in the internal events PRA model, the licensee concluded that the risk from straight-line winds is considered in the RICT calculations.

In Section 6.2.3 of Enclosure 4 of the LAR, the licensee states SSCs whose high wind/tornado-induced failure could prevent safe shutdown of the reactor or result in significant uncontrolled release of radioactivity are protected from such failure because: (1) the structures or components are designed to withstand design wind, tornado wind, and tornado generated missiles, or (2) the system or components are housed within a structure which is designed to withstand the design wind, tornado wind, and tornado generated missiles. Recently, the licensee performed a tornado missile analysis using the Tornado Missile Risk Evaluator (TMRE) in response to NRC Regulatory Issue Summary 2015-06, dated June 10, 2015.[32] The NRC issuance of Amendment Number 169, dated March 29, 2019[33], to utilize TMRE to analyze tornado missile protection nonconformances "concludes that the licensee's evaluation demonstrates that the tornado-missile risk from nonconforming SSCs is acceptably low as it meets the risk acceptance guidelines of RG 1.174."

In its LAR dated February 18, 2019,[34] the licensee reports a high winds CDF of 2.14E-06 per year and LERF of 2.24E-07 per year. The NRC staff notes that these CDF and LERF values are comparable to those reported in the TSTF-505 LAR for internal events and for internal flooding events. Based on the CDF results reported for the High Wind PRA being greater than 1E-06 per year, the NRC staff's concern is that there are potentially high winds vulnerabilities that could impact the RICT calculations. The licensee provided further justification in the supplement dated July 27, 2020, for the screening of the risk of high winds and tornadoes based on the results of a high wind PRA developed for Harris. The licensee provided a breakdown of these risk results into the contributions from hurricanes, straight-line winds, and tornadoes, as follows:

[32] ADAMS Accession No ML15020A419
[33] ADAMS Accession No ML18347A385
[34] ADAMS Accession No ML19049A027

- Hurricanes – CDF contribution of 4.45E-07 per year (20.8 percent) and LERF contribution of 4.66E-08 per year (20.8 percent)

- Straight-line Winds – CDF contribution of 1.38E-06 per year (64.5 percent) and LERF contribution of 1.45E-07 per year (64.7 percent)

- Tornadoes, including from tornado-generated missiles – CDF contribution of 3.23E-07 per year (15.1 percent) and LERF contribution of 3.22E-08 per year (14.4 percent)

With regards to hurricanes, the licensee concluded, as discussed previously, that the risk from hurricanes does not affect the RICT calculations and was screened from inclusion in RICT calculations. For straight-line winds, the licensee determined that 1.03E-06 per year of the CDF contribution and 1.09E-07 per year of the LERF contribution are already accounted for in the internal events PRA. For tornadoes, the licensee determined that 7.62E-08 per year of the CDF contribution and 5.28E-09 per year of the LERF contribution are already accounted for in the internal events PRA. Based on these results, the licensee determined that the portion of the risk from high winds and tornadoes that is not screened or is not already accounted for in the internal events PRA is 5.89E-07 per year for CDF and 6.31E-08 per year for LERF.

The NRC staff finds that the risk from high winds and tornadoes can be excluded from the calculation of RICTs because (1) the risk from hurricane-generated high winds is not applicable to the RICT calculation as the plant will be placed in Mode 3 two hours prior to the anticipated arrival of sustained high winds at the Harris site and the RICT Program cannot be implemented in Mode 3, (2) the applicable CDF contribution from high winds that is not included in the Harris internal events PRA model is less than 1E-06 per year, which meets the PS4 screening criterion of the PRA Standard, and (3) while the PRA Standard does not include explicit screening criterion for LERF, the applicable LERF contribution from high winds that is not included in the Harris internal events PRA model is less than 1E-07 per year, which will have an insignificant impact on RICT calculations.

<u>External Flooding</u>

In Table E4-2 of Enclosure 4 of the LAR, the licensee indicates that its UFSAR and its Plant Evaluation of External Events state that the plant meets the criteria in the 1975 SRP and therefore no safety related structures are impacted because of the maximum still water level or wave run-up from a probable maximum flood, or storm water accumulation due to the probable maximum precipitation. Accordingly, the associated hazard screening criterion (i.e., PS2) from Appendix 6-A of the PRA Standard is indicated as met. In Section 6.3.1 of Enclosure 4 of the LAR, the licensee explained that the Harris external flood hazard was reevaluated as part of the response to the NRC request for information pursuant to 10 CFR 50.54(f) that licensees reevaluate the external flooding hazard at site against the current NRC requirements and guidance Specifically, the adequacy of the plant response to local intense precipitation was evaluated and it was shown that Key Safety Functions are met. A key overall conclusion of the final NRC staff assessment dated December 15, 2017,[35] was that the plant "does not rely on any personnel actions or new modifications to the plant in order to respond to the [local intense precipitation] event." In the NRC staff's assessment, the staff concluded that effective flood protection exists for unbounded mechanisms during a beyond design basis external flooding event at Harris. Given that the revaluation and specifically the evaluation of local intense

[35] ADAMS Accession No. ML17335A121

precipitation did not introduce the need to modify the plant or its operations, the NRC staff finds that the plant continues to meet the criteria in the 1975 SRP.

Other External Hazards

Besides the seismic, extreme winds and tornadoes, and external flooding hazards discussed above, the licensee provided its rationale for concluding there is an insignificant impact of other external hazards for Harris in Table E4-2 of Enclosure 4 to the LAR. The NRC staff's review of the information in the submittal finds that the contributions from the other external hazards have an insignificant contribution to configuration risk and can be excluded from the calculation of the proposed RICTs because they either do not challenge the plant or they are bounded by the external hazards analyzed for the plant.

External Hazards Conclusion

The NRC staff concludes that the licensee's approach for considering the impact of seismic events, and other non-seismic external hazards for Harris in the RICT calculations is acceptable because the licensee (1) included a quantitative bounding assessment of the seismic risk for Harris consistent with the guidance in NEI 06-09, Revision 0-A, that showed it was insignificant to configuration risk, which was confirmed by independent NRC staff calculations, and (2) demonstrated the insignificant contribution to configuration risk from other non-seismic external hazards on the proposed RICTs.

PRA Scope Conclusions

According to the LAR, the proposed RICT Program is only applicable to Operational Conditions (or Modes) 1 and 2, therefore risk evaluations for Modes 3, 4, 5, and 6 are not relevant to the proposed change. Based on its review, the NRC staff finds that the licensee has satisfied the intent and regulatory guidance of RG 1.177, Revision 1 (Section 2.3.2), and RG 1.174, Revision 3 (Sections 2.3 and 2.5), and that the scope of the PRA model and the use of a bounding analysis for seismic events is appropriate for this application.

3.1.4.1.3 *PRA Modeling*

Section 3.2.2 of NEI 06-09-A specifies that to evaluate a RICT for a given Required Action, the specific systems or components involved should be directly modeled in the PRA or, if not directly modeled, the functions directly correlated to the specific systems or components are modeled in the PRA. TSTF-505, Revision 2 also states Required Actions for systems that do not affect CDF or LERF or for which a RICT cannot be quantitatively determined are not in scope of the program. The licensee identified, for each TS LCO Required Actions for which the RICT Program is proposed to apply, the following: (1) the SSCs that are included within the scope of the PRA models, or the surrogate SSCs or operator actions that are modeled that bound the functions of the TS SSCs; and (2) that the PRA success criteria parameters used to determine PRA functional determination are the same as the design basis success criteria parameters or, if different, that plant specific analyses that were used to support the PRA are justified; (3) CCFs are appropriately addressed; and (4) the CRMP provides the capability to select the system as out-of-service in order to calculate a RICT, and the CRMP is maintained consistent with the baseline PRA model.

System and Surrogate Modeling

The NRC staff's SE for NEI 06-09-A specifies that the LAR is to provide a comparison of the TS functions to the PRA modeled functions and that justification should be provided to show that the scope of the PRA model is consistent with the licensing basis assumptions. That SE also specifies that a RICT can be applied to SSCs that are either modeled in the PRA, or whose impact can be quantified using conservative or bounding approaches. In Table E1-1 of Enclosure 1 of the LAR, the licensee identifies each TS within the RICT Program and how the associated systems and components are implicitly or explicitly modeled in the PRA. During its review, the staff determined that the LAR did not provide a sufficient description of the PRA modeling for some systems to allow the NRC staff to understand how a RICT could be applied for certain TSs. The licensee then provided additional details in its supplement, dated July 27, 2020, on the proposed surrogate or bounding modeling to determine the RICTs for several TSs, which is discussed below.

Table E1-1 of Enclosure 1 of the LAR explains that the SSCs for the RTS functional units are not explicitly modeled in the PRA (TS 3.3.1) and so a bounding method for determining the RICTs was presented. The NRC staff noted that some of the SSCs are common to both RTS and the ESFAS, and that ESFAS functional units (TS 3.3.2) are shown to be modeled explicitly in the PRA. Two specific cases are Pressurizer Pressure (Low) and Steam Generator Water Level – Low Low that are part of both TS 3.3.1 and 3.3.2. In some cases, these two systems rely on the same SSCs, yet are treated differently in the calculation of RICTs. The licensee explained in its supplement dated July 27, 2020, that the Pressurizer Pressure and the Steam Generator level instruments applicable to these two TSs, and the associated PRA basic events for each instrument. are modeled explicitly for ESFAS, but not for the associated RTS functions, because the RTS is modeled at a higher level.

For the RTS, the licensee used two surrogate basic events, representing failure of the RTS auto trip function, to determine the RTS contribution to RICTs when these instruments are taken out-of-service. To determine the change in failure probability of these basic events due to out-of-service instruments, the licensee stated it used the original Westinghouse RTS model, which is modeled down to the individual system component level (e.g., instrument channels, trip breakers, etc.). The licensee quantified the respective branches in the Westinghouse model representing the surrogate basic events, and then re-quantified assuming the individual Train A and B RTS trip breaker were out-of-service. The licensee derived a conservative multiplier to apply to the automatic reactor trip initiator. The licensee showed that this multiplier approach produces bounding CDF and LERF values when compared to the detailed Westinghouse model. In calculating the RICT for an instrument that is out-of-service, the initiator multiplier will be applied to model RTS.

The NRC staff concludes the licensee's bounding/surrogate method for determining the RICT when an instrument channel, for the two functions specifically discussed, is removed from service to be reasonable and bounding because: 1) the Westinghouse, NUREG/CR-5500, Volume 2, "Reliability Study: Westinghouse Reactor Protection System 1984-1995," April 1999, RTS fault tree is modeled at a sufficient level of detail to assess the impact on risk from removal of a trip breaker from service, and the component failure data from this NUREG remains applicable today, 2) quantification of the Westinghouse fault tree corresponding to the two surrogate basic events results in failure probabilities that are conservative (or higher) than the corresponding values in the Harris model-of-record, 3) the two surrogate basic events are quantified assuming a trip breaker is out-of-service, which is bounding relative to assuming a single instrument channel for a single RTS parameter is taken out-of-service, and 4) the licensee showed that use of the proposed multiplier results in a higher CDF/LERF, and therefore, a more conservative RICT, than if the failure probabilities for the surrogate basic events were adjusted as described.

Table E1-1 of Enclosure 1 of the LAR, TS 3.3.2 FUNCTIONAL UNIT 1.c Action 19, “Safety Injection (Containment Pressure – High 1),” states that this LCO is modeled with “logically limiting events” that produce a conservatively bounding RICT. A similar description is provided for TS 3.3.2 FUNCTIONAL UNIT 1.e, Action 19, “Safety Injection (Steam Line Pressure – LOW).” In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 2 to identify the logically limiting events used to determine the RICTs for these LCOs and to explain how they are bounding. In its supplement dated July 27, 2020, the licensee explained that the pressure transmitters generating the inputs to the high or low-pressure signals, are not modeled in the PRA. The licensee stated that the associated logic cards are modeled, and this is conservatively bounding. In its RAI response dated November 11, 2020, the licensee, however, removed Functional Unit 1.c from the scope of the RICT program. For Functional Unit 1.e, the NRC staff notes that assuming a logic card out-of-service, instead of assuming a single instrument channel out-of-service is bounding because the ESFAS logic is two out of three positive inputs to initiate safety injection. Accordingly, the NRC staff concludes the licensee’s proposed surrogates for TS 3.3.2 Functional Unit 1.e is conservative and therefore acceptable.

Table E1-1 of Enclosure 1 of the LAR, TS 3.3.2 FUNCTIONAL UNIT 4.c Action 19, “Main Steam Line Isolation (Containment Pressure – High 2)”, states that this LCO is represented by a “bounding surrogate” for the RICT calculation. In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 2 to either remove TS 3.3.2 Function Unit 4.c Action 19 from the RICT Program or provide justification for a different surrogate that is bounding. In its RAI response dated November 11, 2020, the licensee removed TS 3.3.2 Functional Units 4.c and 1.c (Action 19, Safety Injection, Containment Pressure - High 1) from the scope of the RICT Program.

In Note 3 to Table E1-1 of Enclosure 1 of the LAR, the licensee states that “the [Harris] Fire PRA model does not credit containment sprays or containment fan coolers.” Table E1-1 of Enclosure 1 of the LAR proposes to apply the RICT Program to several TS 3.6.2.3, “Containment Spray and Cooling Systems” actions, but it does not address whether bounding or surrogate analyses will be performed for the Fire PRA when determining the RICT for these actions. As a result of the audit conducted by the NRC staff to ascertain the information needed to support its review of the application, the licensee clarified in the supplement dated July 27, 2020, that containment spray and cooling systems are modeled in both the internal events PRA and the Fire PRA. The licensee also provided an overview of the PRA modeling for these systems. Because the licensee confirmed that the subject systems are modeled in both the internal events and the fire PRA, the NRC staff finds a RICT can be estimated for TS 3.6.2.3 and, therefore, the licensee can retain this TS in scope of the RICT Program.

Success Criteria

The NRC staff’s SE for NEI 06-09-A specifies that the LAR is to provide a comparison of the TS functions to the PRA modeled functions and that sufficient justification is to be provided to show that the scope of the PRA model, including applicable success criteria, is consistent with the licensing basis assumptions. Table E1-1 of Enclosure 1 of the LAR identifies each TS within the RICT Program and, as applicable, summarizes how the PRA success criteria differ from the design basis success criteria. In some cases, all the design basis success criteria are not modeled in the PRA (e.g., Containment Spray and Cooling System) or are more restrictive than the PRA success criteria (e.g., the Emergency Core Cooling System success criteria are more restrictive than the PRA success criteria for certain LOCAs). CTs calculated from the PRA will be based on the PRA success criteria which have been reviewed consistent with the PRA technical adequacy review process described in RG 1.200. The use of less restrictive PRA success criteria solely to extend CTs when the design basis criteria can still be satisfied is consistent with NEI 06-09-A, and the associated NRC staff SE and, therefore, is acceptable.

TSTF-505, Revision 2, does not allow for TS loss of function conditions. As stated in Section 2.3 of the TSTF, constraint 1, “Required Actions associated with Conditions that represent a TS loss of specified safety function are outside the scope of the traveler.” The licensee clarified in its supplement dated July 27, 2020, that the definition of loss of safety function, as applied in the LAR, is the same as that defined in TSTF-505, Revision 2, as follows: “a loss of safety function exists when, assuming no concurrent single failure, no concurrent LOOP, or no concurrent loss of onsite diesel generators, a safety function assumed in the accident analysis cannot be performed.”

In Note 3 to Table E1-1 of Enclosure 1 of the LAR, the licensee states that “the [Harris] Fire PRA model does not credit containment sprays or containment fan coolers.” Table E1-1 of Enclosure 1 of the LAR proposes to apply the RICT program to several TS 3.6.2.3, “Containment Spray and Cooling Systems,” actions but does not address whether bounding or surrogate analyses will be performed for the Fire PRA when determining the RICT for these actions. The licensee clarified in its supplement dated July 27, 2020, that the Containment Fan Coolers and the Containment Spray System are redundant for performing the safety function to cool the containment atmosphere following a severe accident.

Table E1-1 of Enclosure 1 of the LAR, TS 3.7.1.5 ACTION for MODE 1, specifies the design basis success criteria for steam line rupture. The NRC staff is unclear how an inoperable and open MSIV may affect the safety analysis for the steam generator tube ruptures event and whether this condition would represent a TS loss of function within this context (i.e., to isolate the ruptured steam generator). As a result of the audit conducted by the NRC staff to ascertain the information needed to support its review of the application, the licensee further explained in the supplement dated July 27, 2020, that the safety function of the MSIVs is to prevent the uncontrolled blowdown of more than one steam generator following a steam line break and that capability is maintained with one MSIV inoperable (but open). The licensee further explained that MSIVs do not perform a TS specified safety function with respect to a SGTR event.

Table E1-1 of Enclosure 1 of the LAR, TS 3.3.2 Functional Unit 6.f Action 15 Auxiliary Feedwater (Trip of all Main Feedwater Pumps), specifies that the design success criteria are one channel per pump. Table 3.3-3 of the TS, as provided in Attachment 2 of the LAR, specifies that the total number of channels per pump is one. The licensee indicated in its supplement dated July 27, 2020, that this TS action is being removed from the scope of the RICT Program because it represents a loss of function condition.

<u>Common-Cause Failure (CCF) Modeling</u>

Section 3.3.6 of NEI 06-09-A states that for all RICT assessments of planned configurations, the treatment of CCF may be performed by considering only the removal of the out-of-service equipment and not adjusting CCF terms. RG 1.177 states that when a component is rendered inoperable in order to perform preventative maintenance, the CCF contributions in the remaining

operable components should be modified to remove the inoperable component and to only include CCF of the remaining components.

The LAR does not specifically address how the potential for CCFs will be accounted for in the RICT calculations. The licensee explained in its supplement dated July 27, 2020, that common cause basic events are explicitly modeled in the PRA, with each specific combination of events modeled in conjunction with the independent failure basic event. The licensee stated that common cause basic events are not adjusted in the PRA models when a component is taken out-of-service for planned maintenance. The licensee further explained that this approach is conservative because the CCF basic events covering the specific combinations of CCF are retained for components removed from service. The NRC staff notes that this simplification of not adjusting the CCFs for an out-of-service component can also produce both conservative and non-conservative effects. However, the CCF probability estimates are uncertain, and retaining precision in calculations using refined probabilities will not necessarily improve the accuracy of the results. Therefore, the NRC staff concludes that the licensee's general treatment of CCFs for planned maintenance is acceptable because the calculations reasonably include CCFs after removing one train for maintenance consistent with the accuracy of the estimates.

RG 1.177 also states that when a component is rendered inoperable due to a failure, the CCF probability for the remaining redundant components should be increased to represent the conditional failure probability due to CCF of these components, in order to account for the possibility that the first failure was caused by a CCF mechanism.

In Attachment 2 of the LAR, the licensee proposed to include a new administrative TS requirement (i.e., constraint r. of TS Section 6.8) defining the limitations on implementation of the RICT Program. The new TS specifies that in an emergent condition, if the extent of condition for the inoperable SSC is not complete prior to exceeding the completion time, then the RICT Program will account for the increased possibility of CCF by either numerically accounting for the increased possibility of CCF in the RICT calculation or implementing RMAs not already credited in the RICT calculation that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

The licensee clarified in its supplement dated July 27, 2020, that the numerical adjustment of CCF events would not typically be performed for a RICT calculation, but that RMAs would generally be put in place, in accordance with TS Section 6.8, constraint r, item d, if CCF cannot be ruled out. The licensee also confirmed that, if a numerical adjustment is performed, the guidance in Section A-1.3.2.1 of Appendix A of RG 1.177, Revision 1, would be followed in making this adjustment. The NRC staff finds that the licensee's approach to addressing CCF in an emergent condition is acceptable because it is in accordance with the regulatory guidance in RG 1.177 when using option 1 of TS Section 6.8, constraint r, item d.

<u>CRMP Model</u>

The PRA model serves as the model used by the CRMP tool, which is used to perform the RICT calculations. The Harris CRMP tool is referred to in Enclosure 8 of the LAR as the RTR tool. The RTR tool is used to evaluate the PRA models (i.e., the internal events, internal flooding events, and fire events PRA models) with a zero-maintenance baseline and the actual plant configuration. In order to reflect the current plant configuration rather than the average plant configuration represented in the baseline PRA models, the RTR model includes adjustments made to the PRA models. These adjustments are described in Enclosure 8 of the LAR. The licensee further explained that the PRA models may be optimized to improve performance speed, but that these changes are verified to provide the same results as the baseline models. The unit specific configuration results from the internal events PRA, internal flooding events PRA, and fire events PRA are quantified separately and combined with the fixed seismic CDF/LERF contributions in the RTR software when calculating a RICT or RMA time. The tool used to perform the RICT calculations provides a user interface which supports the RICT Program by providing a method to evaluate the plant configuration.

As a result of the audit conducted by the NRC staff to ascertain the information needed to support its review of the application, the licensee further explained in the supplement dated July 27, 2020, that two PRA-modeled systems must account for seasonal variations: EDG heating, ventilation, and air conditioning, which requires two fans for summer operation and one fan for winter operation, and Switchgear Rooms which require ventilation during winter operation. The licensee indicated that these adjustments are made in the RTR tool. The licensee also stated that out-of-service equipment is properly reflected in the RTR model initiating event fault trees, as well as plant response fault trees. The pre-solved cutsets are not used for making RICT calculations. The licensee described the benchmarking activities performed to confirm consistency of the RTR model results to the PRA Models of Record results.

The NRC staff concludes that the licensee appropriately accounts for seasonal variations in its RTR tool and that it has a process for verifying that the CRMP (or RTR) model used to calculate the RICTs is consistent with the underlying baseline PRA. The CRMP model used to calculate the RICTs is acceptable because the underlying PRA models will remain acceptable and the licensee will verify, through its benchmarking activities, that the CRMP model is consistent with the underlying baseline PRA.

PRA Modeling Conclusions

The NRC staff reviewed the information provided by the licensee and concluded that the PRA modeling used to support the RICT Program can appropriately model alignments of components during periods when the RICT will be calculated. Therefore, the NRC staff finds that the licensee has satisfied the intent and regulatory guidance of RG 1.177, Revision 1 (Section 2.3.3), and RG 1.174, Revision 3 (Section 2.3), and that the PRA modeling is appropriate for the application of the RICT Program.

3.1.4.1.4 *Assumptions, Sensitivity, and Uncertainty Analyses*

According to RG 1.177, Revision 1, using PRAs to evaluate TS changes requires consideration of the assumptions made within the PRA that can have a significant influence on the ultimate acceptability of the proposed changes. Risk-informed analyses of TS changes can be affected by uncertainties regarding the assumptions made during the PRA model's development and application. Typically, the risk resulting from TS CT changes is expected to be relatively insensitive to most uncertainties because the uncertainties tend to affect similarly both the pre-TS change case and the post-TS change case. The NRC staff's SE for NEI 06-09-A specifies that the LAR is to provide a discussion of how the PRA model key assumptions and sources of uncertainty were identified, and how their impact on the RMTS was assessed and dispositioned.

Evaluation of PRA Assumptions and Sources of Uncertainty

In Enclosure 9 of its submittal, the licensee discussed its process for determining the assumptions and sources of uncertainty for the PRA models and for determining which of those are key for the application. The licensee also identified and dispositioned each of the key assumptions and sources of uncertainty for its impact on the RICT calculations. The licensee evaluated each of the Harris PRA models (i.e., internal events, internal flooding events, and internal fire events) to identify the associated assumptions and sources of uncertainty. According to the LAR, the licensee followed the guidance in NUREG-1855 to identify assumptions and sources of uncertainty in each of the PRA models, including (1) reviewing the generic issues identified in EPRI Report 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," dated December 2008, for internal events and internal flooding events, (2) reviewing the generic issues identified in EPRI Report 1026511, "Practical Guidance of the Use of Probabilistic Risk Assessment in Risk-informed Applications with a Focus on the Treatment of Uncertainty," for internal fire events, and (3) reviewing the Harris PRA documentation for plant-specific assumptions and uncertainties. NUREG-1855, Revision 1, cites EPRI Report 1016737 and EPRI Report 1026511 as providing specific methods for identifying sources of PRA modeling uncertainty.

The licensee evaluated each identified assumption and source of uncertainty against definitions and guidance for "key" specified in RG 1.200, Revision 2, NUREG-1855, Revision 1, and EPRI Report 1013491, "Guideline for the Treatment of Uncertainty in Risk-Informed Applications," dated October 2006. Any assumption or source of uncertainty that did not meet these definitions were screened by the licensee as not being key assumptions or uncertainties. Based on these definitions, the following considerations were used to screen assumptions and sources of uncertainty as not key to the RICT application:

- The uncertainty or assumption is implementing a consensus model.
- The uncertainty or assumption has no impact on the PRA results.
- There is no reasonable alternative to the assumption.
- The uncertainty or assumption implements a slight realistic conservatism that does not influence the PRA results.
- The uncertainty or assumption implements a model that has extensive historical precedence.

Table E9-1 of Enclosure 9 of the LAR, as supplemented by the letter dated July 27, 2020, presented the assumptions and sources of uncertainty determined by the licensee to be potentially key to the RICT application and their associated disposition. The licensee determined that none of the key assumptions and uncertainties would impact the RICT calculations. The NRC staff reviewed the licensee's dispositions of the assumptions it had identified to be potentially key to the RICT application and found the licensee's dispositions acceptable. However, it is unclear from the LAR whether the sources of modeling uncertainty (both plant-specific and generic) were identified for the LERF PRA model. In an email dated September 29, 2020, the NRC staff asked the licensee in RAI 3 to describe the process used to identify and evaluate key assumptions and sources of model uncertainty for the Level 2 PRA models. In addition, for any identified key assumptions and sources of model uncertainty, the

NRC staff requested the licensee to provide justification that the impacts of these key uncertainties and assumptions on the RICT Program are acceptable.

In its RAI 3 response dated November 11, 2020, the licensee provided the results of an assessment of the generic sources of modeling uncertainties in the Level 2 PRA, which are identified in EPRI 1026511. The licensee explained that this assessment was performed in accordance with NUREG-1855 and with the process described in the LAR. The licensee stated it reviewed the PRA for model updates performed since submittal of the LAR and no additional key assumptions or sources of uncertainty were identified. Regarding the assessment process used by the licensee, the licensee explained that, in addition to the considerations identified in the LAR for identifying key assumptions and sources of uncertainty, the assessment also considered whether the assumption is a statement of fact or if it does not pertain to the application (e.g., a Level 2 PRA issue that does not impact LERF). The determination of whether an assumption was a key assumption or source of uncertainty was a judgement by an experienced PRA analyst based on their assessment of its importance to the overall PRA model, the RICT application, and how the PRA model is utilized for the RICT application. The NRC staff finds that the licensee's process for the identification and assessment of Level 2 PRA modeling uncertainties is reasonable because it follows the guidance in NUREG-1855 and considered both generic and plant-specific assumptions and sources of uncertainty.

<u>Credit for FLEX</u>

The NRC staff's memorandum, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-informed Decision Making,' Guidance for Risk-informed Changes to Plants Licensing Basis," dated May 30, 2017,[36] provides the NRC staff's assessment of challenges to incorporating diverse and flexible coping strategies (FLEX) equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance in RG 1.200, Revision 2. To address this concern, the licensee stated in its supplement dated July 27, 2020, that credit is taken for FLEX equipment in the internal events PRA and internal flooding PRA models only for extended loss of AC power scenarios. The licensee further explained in its RAI 4 response dated November 11, 2020, that the credit includes permanently installed diesel generators and portable AFW pumps. The licensee stated that because there is no failure rate data for the FLEX equipment, it used the failure data for EDG and permanently installed diesel pumps compiled in NUREG/CR-6928, 2015 Update, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," dated December 2016.[37]

The licensee indicated that credited FLEX operator actions include failure to load shed, failure to align and start the FLEX DG, failure to refuel the FLEX DG, and failure to align and start the FLEX AFW pump. The licensee stated in its supplement dated July 27, 2020, that these FLEX actions are included in the activities described in Sections 7.5.4 and 7.5.5 of NEI 16-06, "Transmittal of NEI 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,'" dated August 26, 2016,[38] for which current human reliability analysis methods may not be applicable. The human error probabilities for each operator action were developed using the same PRA methodology as used in the peer reviewed PRA models and information regarding training frequency and timing validation for actions. The licensee provided a summary of the timing and performance shaping factors, as identified in supporting requirements HR-G3 of the

[36] ADAMS Accession No. ML17031A269
[37] https://nrcoe.inl.gov/resultsdb/publicdocs/AvgPerf/ComponentUR2015.pdf
[38] ADAMS Accession No. ML16286A297

PRA Standard, used in the human reliability analysis to develop the operator failure probabilities for each FLEX human failure event. The licensee assessed that the FLEX human failure events are conservative based on: (1) the significant time margin (at least three hours) between the available time to complete each action before the onset of core damage and the estimated time to detect and diagnose an extended loss of AC power, make the decision to implement the FLEX strategies, and execute the required response, (2) the low complexity associated with detecting and diagnosing that an extended loss of AC power (LOOP and failure of the EDGs) has occurred, and making the decision to execute FLEX strategies per plant procedures because there are no other options, and (3) the assumption that execution stress levels are high. The licensee also reported that there are two human failure event combinations to which a floor value of 1E-06 was applied and justified that this was an appropriate floor based on the long-time window (21 hours) to complete the actions.

The NRC staff's memorandum dated May 30, 2017, highlights two main areas of uncertainties for crediting FLEX in the PRA: equipment failure probabilities and human reliability analysis of the credited operator actions for deploying FLEX. The guidance in NEI 06-09-A states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT Program on uncertainties that could potentially impact the results of the RICT calculation, and that the insights from the sensitivity studies should be used to develop appropriate compensatory RMAs, including highlighting risk-significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. The licensee provided in its supplement dated July 27, 2020, the results of sensitivity studies, where credit for FLEX was removed in order to assess the impact on calculated RICTs. Sensitivity study results were provided for TS 3.8.1.1 Action c.1 in which one offsite circuit and one emergency DG are inoperable, and TS 3.8.2.1 Action for two cases: (1) one battery not available and (2) one DC bus not available. In each case the RICT was shown to be insignificantly impacted by crediting FLEX. These results reflect impact on the internal events PRA model. The licensee explained that the internal flooding PRA model is not sensitive to FLEX and the fire PRA does not credit FLEX.

Based on its review, the NRC staff verified that, the licensee's sensitivity analysis results that showed crediting FLEX in the internal events PRA model for extended loss of AC power scenarios has a minor impact on calculated RICTs. The NRC staff finds that crediting FLEX for extended loss of AC power scenarios in the internal events PRA model is acceptable for use in the Harris RICT Program. Future plant and industry operating experience affecting the FLEX equipment failure rates and HEPs is expected to be captured by the PRA maintenance process. As described in Section 3.1.4.1.6 of this SE, consistent with the guidance in NEI 06-09-A, the licensee has developed procedures for maintaining the PRA to reasonably reflect the as-built and as-operated plant. In the future, if FLEX actions are identified that are important for a RICT, the guidance in NEI 06-09-A, directs licensees to use configuration-specific RMAs on a case by case basis, as needed, to address the impact of uncertainties.

Digital I&C

The NRC staff noted that the lack of consensus industry guidance for modeling digital I&C systems in plant PRAs is a source of uncertainty. The licensee explained in its supplement dated July 27, 2020, that the only digital components modeled in the Harris PRA are the RTS logic cards for the Westinghouse Solid State Protection System. The licensee further explained in its RAI 7 response dated November 11, 2020, that the reliability data for these components is obtained from NUREG/CR-6928. The licensee performed a sensitivity study that increased the failure probabilities of these logic cards by a factor of three. The licensee stated that the results of this sensitivity analysis showed no impact on the RICTs for impacted LCOs.

Assumptions, Sensitivity, and Uncertainty Analyses Conclusions

The NRC staff's review indicates the licensee performed an adequate assessment of potential sources of PRA uncertainty, and the identification of the key assumptions and sources of uncertainty was appropriate and consistent with the guidance in NUREG-1855, EPRI Report 1016737 and EPRI Report-1026511, and associated regulatory requirements and guidance. Therefore, the NRC staff finds that the licensee has satisfied the guidance in RG 1.177, Revision 1, and RG 1.174, Revision 3, and that the identification of assumptions and treatment of model uncertainties for risk evaluation of extended CTs is appropriate for this application and consistent with the guidance identified in NEI 06-09-A.

3.1.4.1.5 *PRA Results and Insights*

The proposed LAR implements a process to determine TS RICTs rather than specific changes to individual TS CTs. Topical Report NEI 06-09-A requires periodic assessment of the risk incurred due to operation beyond the "front stop" CTs due to implementation of a RICT Program and comparison to the guidance of RG 1.174, Revision 3, for small increases in risk. As with other unique risk-informed applications, supplemental risk acceptance guidelines that complement the RG 1.174 guidance are appropriate. NEI 06-09-A instructs that configuration risk should be assessed to determine the RICT and establishes the criteria for ICDP and ILERP on which to base the RICT. An ICDP of 1E-5 and an ILERP of 1E-6 are used as the risk measures for calculating individual RICTs. These limits are consistent with NUMARC 93-01. The use of these limits in NEI 06-09-A aligns the TS CTs with the risk management guidance used to support plant programs for the Maintenance Rule, and the NRC staff accepted these supplemental risk acceptance guidelines for RMTS programs in its approval of NEI 06-09-A.

Topical Report NEI 06-09-A as modified by the limitations and conditions in the associated SE, instructs that the cumulative impact of implementation of an RMTS be periodically assessed and shown to result in: (1) a total risk impact below 1E-5/year for changes to CDF, (2) a total risk impact below 1E-6/year for changes to LERF, and (3) the total CDF and total LERF must be reasonably shown to be less than 1E-4/year and 1E-5/year, respectively. The licensee indicated in Enclosure 5 of the submittal that the estimated total CDF and LERF meet the 1E-4/year CDF and 1E-5/year LERF criteria of RG 1.174 consistent with the guidance in NEI 06-09-A and that these guidelines will be satisfied whenever a RICT is implemented.

The licensee has incorporated NEI 06-09-A in the RICT Program of TS 6.8.4.r, and therefore, can calculate the RICT consistently with its criteria and assesses the RICT Program to assure that any risk increases are small per the guidance of RG 1.174, Revision 3, and the intent of RG 1.177, Revision 1. Also, the licensee's estimates of the current total CDF and LERF meet the RG 1.174, Revision 3 guidelines. Therefore, the NRC staff finds that the licensee's RICT Program is consistent with NEI 06-09-A guidance and is acceptable.

3.1.4.1.6 *Implementation of the RICT Program*

Because NEI 06-09-A involves the real-time application of PRA results and insights by the licensee, the NRC staff reviewed the licensee's description of programs and procedures associated with implementation of the RICT Program in Enclosure 10 of its submittal. The administrative controls on the PRA and on changes to the PRA should provide confidence that the PRA results are reasonable, and the administrative controls on the plant personnel using the RICT should provide confidence that the RICT Program will be appropriately applied.

The means for demonstrating the technical acceptability of the PRA models include assessment against the PRA standards and RG 1.200, which includes guidance for performing peer reviews and focused-scope peer reviews. The technical adequacy of the PRA models is discussed in Enclosure 2 and Enclosure 7 of the submittal. According to Enclosure 8 of the submittal, future changes made to the baseline PRA model, changes made to the baseline PRA model for translation to the online model, and changes made to the online model configuration files, are controlled and documented by plant procedures.

Topical Report NEI 06-09-A specifies that the RMTS risk assessment process should be integrated into station-wide work control processes and defines the necessary attributes of the RMTS Program structure. In the conduct of RMTS, procedural guidance is needed for conducting and using the results of the risk assessment. These procedures should specify the station functional organizations and personnel, including operations, engineering, work management and PRA personnel, responsible for each step of the procedures. The procedures should also clearly specify the process for calculating the applicable RICT, implementing RMAs, conducting, reviewing, and approving decisions to exceed the front-stop CT and remove equipment from service.

Enclosure 10 of the LAR describes the implementing programs and procedures and the associated personnel training. The licensee explained that a RICT Program description and implementing procedures will be developed. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT Program. More detailed procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT Program. The program description and implementing procedures will incorporate the programmatic guidance for RMTS included in NEI 06-09-A. The program will be integrated with the existing Harris online work control process. Entry into the RICT Program will require management approval prior to pre-planned activities and as soon as practicable following emergent conditions. These and other attributes that will be addressed in the RICT Program are identified in the LAR. Because the licensee's procedures will be developed using the guidance in NEI-06-09, and will be subject to inspection by the NRC staff, the staff concludes that the licensee's proposed development of appropriate implementing programs and procedures is acceptable.

The NRC staff concludes that the licensee will establish appropriate programmatic and procedural controls for its RICT Program, consistent with the guidance of NEI 06-09-A Section 3.2.1. Topical Report NEI 06-09-A specifies that stations implementing an RMTS program shall provide training, in the programmatic guidance associated with the RMTS program and of the individual RICT evaluations, to personnel responsible for determining TS operability decisions or conducting RICT assessments. NEI 06-09-A further specifies that training of plant personnel should be provided for organizations with functional responsibilities for performing or administering the CRMP commensurate with each position's responsibilities, in accordance with 10 CFR 50.120(b)(3) and other applicable regulations, within the RICT Program, as described in NEI 06-09-A. In furtherance of its adherence to this guidance, the licensee identified the categories of plant personnel that will be trained and the different types of training that the different categories of plant personnel receive.

The NRC staff reviewed the description of the training program provided in the LAR, and concluded that the program is consistent with the training guidance set forth in NEI 06-09-A.

Therefore, the NRC staff finds that the licensee has proposed acceptable administrative controls on procedures and training for the RICT Program.

3.1.4.2 Tier 2: Avoidance of Risk-Significant Plant Configurations

The second tier provides that a licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out-of-service in accordance with the proposed TS change. Topical Report NEI 06-09-A would not permit voluntary entry into high-risk configurations, which would exceed instantaneous CDF and LERF limits of 1E-3/year and 1E-4/year, respectively. The guidance in NEI 06-09-A specifies that if the instantaneous CDF and LERF limits are exceeded for emergent conditions, then implementation of RMAs is needed. It further specifies the need for implementation of RMAs when the actual or anticipated risk accumulation during a RICT will exceed one-tenth of the ICDP or ILERP limit (the RMA time). Such RMAs may include rescheduling planned activities to lower risk periods or implementing risk-reduction measures. The RICT Program requirements and criteria are consistent with the principle of Tier 2 to avoid risk-significant configurations.

Consistent with NEI 06-09-A, Enclosure 12 of the LAR identifies three categories of RMAs (i.e., actions to provide increased risk awareness and control, actions to reduce the duration of maintenance activities, and actions to minimize the magnitude of the risk increase). The LAR also explains that RMAs will be implemented, in accordance with plant procedures, (1) no later than the time at which the 1E-06 ICDP or 1E-07 ILERP thresholds are reached, (2) under emergent conditions when the instantaneous CDF or LERF thresholds are exceeded, and (3) under emergent conditions, if the extent of condition is not known prior to exceeding the CT, to account for the increased possibility of CCF (see Section 3.1.4.1.3 of this SE). Additionally, if an emergent event occurs in which an existing RMA time has been exceeded, the RMAs already implemented will be reevaluated to determine if new RMAs are appropriate. Enclosure 12 of the LAR also provides several examples of RMAs for several selected LCO conditions.

The licensee further explained in its supplement dated July 27, 2020, that determination of RMAs is performed using plant procedures and involves both qualitative and quantitative considerations based on configuration-specific risk. The development of RMAs is performed in a graded manner based on risk levels, with increasing risk levels prescribing more RMAs with increased scope to, for example, protect equipment, brief operators, etc. If emergent issues cause entry into a high risk (severely degraded) condition, which cannot be entered voluntarily, immediate actions are taken to reduce risk including restoring components important to accident mitigation to, at least, a functional state. If risk cannot be reduced in a reasonable amount of time, an orderly transition to Mode 3 is considered.

For configurations that may be susceptible to CCFs, the licensee explained that, per plant procedure, common cause RMAs lower configuration risk by focusing on: (1) availability of SSCs providing redundancy to the failed SSC, (2) availability of diverse SSCs providing redundancy for functions performed by the failed SSC, (3) reducing the likelihood of initiating events that can impact the availability of the redundant or diverse SSCs, (4) ensuring the readiness of operators to respond to initiating events assuming SSCs susceptible to failure by common cause will fail, and (5) ensuring the readiness of maintenance personnel to respond to additional failures of redundant or diverse SSCs. Examples of qualitatively determined RMAs, quantitatively determined RMAs and common cause RMAs are provided in the supplement dated July 27, 2020. The NRC staff concludes that the licensee's process for developing RMAs is in accordance with NEI 06-09-A because it utilizes configuration-specific risk insights and specifically considers the potential for CCFs in emergent conditions.

The licensee clarified in its supplement dated July 27, 2020, that it will follow the guidance in NEI 06-09-A, in that the use of the RICT Program is precluded for planned equipment outages if the instantaneous CDF or LERF are higher than the limits of 1E-03 or 1E-04, respectively. Based on the licensee's incorporation of NEI 06-09-A in the TS as discussed in Attachment 1 of the LAR and use of RMAs as discussed in Enclosure 12 of the LAR and the supplemental information, and because the proposed changes are consistent with the guidance of RG 1.174, Revision 3, and RG 1.177, Revision 1, the NRC staff finds the licensee's Tier 2 program is acceptable and supports the proposed implementation of the RICT Program.

3.1.4.3 Tier 3: Risk-Informed Configuration Risk Management

The third tier provides that a licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.

Topical Report NEI 06-09 Revision 0-A addresses Tier 3 guidance by specifying that assessment of the RICT is to be based on the plant configuration of all SSCs that might impact the RICT, including safety-related and non-safety-related SSCs. If a risk-significant plant configuration exists, based on the expectation of exceeding a threshold of one-tenth of the risk on which the RICT is based, compensatory measures and RMAs are required to be implemented. Therefore, the NRC staff finds that the RICT Program provides an acceptable methodology to assess and address risk-significant configurations. Consistent with NEI 06-09-A, a reassessment of any plant configuration changes will need to be completed in a timely manner based on the more restrictive limit of any applicable TS action requirement or a maximum of 12 hours after the configuration change occurs.

Based on the licensee's incorporation of NEI 06-09-A in the TS, as discussed in Attachment 1 of the LAR and use of RMAs as discussed in Enclosure 12 of the LAR, as supplemented, and because the proposed changes are consistent with the Tier 3 guidance of RG 1.177, Revision 1, the NRC staff finds that the proposed changes are acceptable.

3.1.4.4 Key Principle 4 Conclusions

The licensee has demonstrated the technical acceptability and scope of its PRA models, and that the models can support implementation of the RICT Program for determining CTs. Proper consideration of key assumptions and sources of uncertainty have been made. The risk metrics are consistent with the approved methodology of NEI 06-09-A and the acceptance guidance in RG 1.177 and RG 1.174. The RICT Program is controlled administratively through plant procedures and training. The RICT Program follows the NRC-approved methodology in NEI 06-09-A. The NRC staff concludes that the RICT Program satisfies the fourth key safety principle of RG 1.177 and is, therefore, acceptable.

3.1.5 Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring Program

Revision 1 of RG 1.177 and RG 1.174, Revision 3, establish the need for an implementation and monitoring program to ensure that extensions to TS CTs do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. Revision 3 of RG 1.174 states that monitoring performed in conformance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed is sufficient for the SSCs affected by the risk-informed application. According to Enclosure 11 of the LAR, the SSCs in the scope of the RICT Program are also in the scope of the Maintenance Rule. The Maintenance Rule monitoring programs will provide for evaluation and disposition of unavailability impacts which will may be incurred from implementation of the RICT Program.

Section 3.3.3 of NEI 06-09-A instructs the licensee to track the risk associated with all entries beyond the "front stop" CT, and Section 2.3.1 provides a requirement for assessing cumulative risk, including a periodic evaluation of any increase in risk due to the use of the RMTS Program to extend the CTs. According to Enclosure 11 of its submittal, the licensee calculates cumulative risk at least every refueling cycle, but the recalculation period does not exceed 24 months, which is consistent with NEI 06-09-A. The licensee converts the cumulative ICDP and the ILERP into average annual values which are then compared to the limits of RG 1.174. If any acceptance guidelines are exceeded, corrective actions are taken to ensure that future plant operational risk is within the acceptance guidelines. This evaluation assures that RMTS Program implementation meets RG 1.174 guidance for small risk increases.

3.1.5.1 Key Principle 5 Conclusions

The NRC staff concludes that the RICT Program satisfies the fifth key safety principle of RG 1.177, Revision 1, and RG 1.174, Revision 3, by, in part, monitoring the average annual cumulative risk increase as described in NEI 06-09-A and using this average annual increase to ensure the program as implemented meets the RG 1.174 guidance for small risk increases and, therefore, is acceptable. Additionally, the NRC staff concludes that the RICT Program satisfies the fifth key safety principle of RG 1.177, Revision 1, and RG 1.174, Revision 3, because, in part, all the affected SSCs are within the Maintenance Rule Program which can be used to monitor changes to the reliability and availability of these SSCs.

3.2 VARIATIONS FROM TSTF-505

The NRC staff evaluated the proposed use of RICTs in the variations stated above in Section 2.2.3 in conjunction with evaluating the proposed use of RICTs in each of the individual LCO actions and CTs stated above in Section 2.2.2. The NRC staff's evaluation of the licensee's proposed use of RICTs in the variations against the key safety principles is discussed above in Sections 3.1.1 through 3.1.5. Based on the above Sections 3.1.1 through 3.1.5, the NRC staff finds that each of the five key principles in RG 1.177, Revision 1, and RG 1.174, Revision 3, have been met and concludes that the proposed variations are acceptable.

3.3 TECHNICAL SPECIFICATION ADMINISTRATIVE CONTROLS SECTION

The NRC staff reviewed the licensee's proposed addition of a new program, the RICT Program, to the Administrative Controls section of the TS. The NRC staff evaluated the elements of the new program to ensure alignment with the requirements in 10 CFR 50.36(c)(5) and to ensure the programmatic controls are consistent with the RICT Program described in NEI 06-09-A. Technical Specification 6.8.4.r requires that the RICT Program be implemented in accordance with NEI 06-09-A. This is acceptable because NEI 06-09-A establishes an appropriate framework for an acceptable RICT Program. The TS states that a RICT may not exceed

30 days. The NRC staff determined that 30-day limit is appropriate because it allows sufficient time to restore SSCs to operable status while avoiding excessive out-of-service times for TS SSCs. The TS states that the RICT may only be used in Mode 1 and 2. This provision ensures that the RICT is only used for determination of CDF and LERF for modes of operation modeled in the PRA. The TS requires that while in a RICT, any change in plant configuration as defined in NEI 06-09-A must be considered for the effect on the RICT. The TS also specifies time limits for determining the effect on the RICT. These time limitations are consistent with those specified in NEI 06-09-A.

The TS contains requirements for the treatment of CCFs for emergent conditions in which the common cause evaluation is not complete. The requirements are to either (a) numerically account for the increased probability of CCF, or (b) to implement RMAs that support redundant or diverse SSCs that perform the functions of the inoperable SSCs and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs. Key Principle 2 of risk-informed decision-making is to assure that the change is consistent with defense-in-depth philosophy. The seven considerations supporting the evaluation of the impact of the change on defense-in-depth are discussed in RG 1.174, including one to preserve adequate defense against potential CCF. The NRC staff finds that numerically accounting for an increased probability of failure will shorten the estimated RICT based on the particular SSCs involved thereby limiting the time when a CCF could affect risk. Alternatively, implementing actions that can increase the availability of other mitigating SSCs or decrease the frequency of demand on the affected SSCs will decrease the likelihood that a CCF could affect risk. The NRC staff concludes that both the quantitative and the qualitative actions minimize the impact of CCF and therefore, support meeting Key Principle 2 as described in RG 1.174. These methods either limit the exposure time, help ensure the availability of alternate SSCs, or decrease the probability of plant conditions requiring the safety function to be performed. The NRC staff finds that these methods contribute to maintaining defense-in-depth because the methods limit the exposure time or ensure the availability of alternate SSCs.

The TS contains a provision that risk assessment approaches and methods used shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in RG 1.200, Revision 2. Methods to assess the risk from extending the CTs must be PRA methods used to support this LAR, or other methods approved by the NRC for generic use. As stated in the NRC staff's SE for NEI 06-09-A:

> TR NEI 06-09, Revision 0, requires an evaluation of the PRA model used to support the RMTS against the requirements of RG 1.200, Revision 1, and ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", for capability Category II. This assures that the PRA model is technically adequate for use in the assessment of configuration risk. This capability category of PRA is sufficient to support the evaluation of risk associated with out of service SSCs and establishing risk-informed CTs.

Technical Specification 6.8.4.r was updated to reflect the current revision of RG 1.200. RG 1.200 incorporates ASME RA-S-2002 by reference.

The NRC staff's SE for NEI 06-09-A also states:

> As part of its review and approval of a licensee's application requesting to implement the RMTS, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods approved by the NRC staff for use in the plant-specific RMTS program. If a licensee wishes to change its methods, and the change is outside the bounds of the license condition, the licensee will need NRC approval, via a license amendment, of the implementation of the new method in its RMTS program. The focus of the NRC staff's review and approval will be on the technical adequacy of the methodology and analyses relied upon for the RMTS application.

This limitation and condition is being relocated from a license condition to the Administrative Controls section of the TS. Proposed TS 6.8.4.r restates this limitation and condition from the NRC staff's SE in language that is appropriate for the Administrative Controls section of the Harris TS. This constraint appropriately requires the licensee to utilize the risk assessment approaches and methods previously approved by the NRC and/or incorporated in the RICT Program, and requires prior NRC approval for any change in PRA methods to assess risk that are outside those approval boundaries. The NRC staff finds that this requirement is appropriately reflected in the Administrative Controls section of the Harris TS.

The regulations in 10 CFR 50.36(c)(5) require the TS to contain administrative controls providing "provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." The NRC staff has determined that Administrative Controls section of the TS will assure operation of the facility in a safe manner when the facility uses the RICT Program. Therefore, the NRC staff has determined that the requirements of 10 CFR 50.36(c)(5) are satisfied.

4.0 SUMMARY

The NRC staff finds that the licensee's proposed implementation of the RICT Program for the identified scope of Required Actions is consistent with the guidance of NEI 06-09-A. The licensee's methodology for assessing the risk impact of extended CTs, including the individual CT extension impacts in terms of ICDP and ILERP, and the overall program impact in terms of ΔCDF and ΔLERF, is accomplished using PRA models of sufficient scope and technical adequacy based on consistency with the guidance of RG 1.200, Revision 2. For seismic hazards, which do not have a PRA model, the licensee will use bounding analyses in accordance with NEI 06-09-A guidance and Administrative Control TS. The RICT calculation uses the PRA model as translated into the CRMP tool, and the licensee has an acceptable process in place to ensure the quality of the translation. In addition, the NRC staff finds that the proposed implementation of the RICT Program addresses the RG 1.177 defense-in-depth philosophy and safety margins to ensure that they are adequately maintained and includes adequate administrative controls as well as performance monitoring programs.

The regulation at 10 CFR 50.36(a)(1) states, in part: "[a] summary statement of the bases or reasons for such specifications other than those covering administrative controls shall also be included in the application, but shall not become part of the technical specifications." Accordingly, along with the proposed TS changes, the licensee also submitted TS Bases changes that correspond to the proposed TS changes, to provide the reasons for those TSs. The NRC staff found the TS Bases changes to be consistent with the bases changes in the model application dated July 2, 2018.[39]

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment on February 8, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (85 FR 2160), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 2, 2021

[39] ADAMS Accession No. ML18183A493

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 184 REGARDING TECHNICAL SPECIFICATIONS TASK FORCE (TSTF) TRAVELER TSTF-505, REVISION 2, “PROVIDE RISK INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4B,” (EPID L-2019-LLA-0218) DATED APRIL 2, 2021

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RidsACRS_MailCTR
RidsNrrPMShearonHarris
RidsRgn2MailCenter
RidsNrrDraAplc
RidsNrrDraApla
RidsNrrDssStsb
RidsNrrDssScpb
RidsNrrDssSnsb
RidsNrrDexEicb
RidsNrrDexEeob
RidsNrrDexEmib

ADAMS Accession No.: ML21047A314

OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DSS/SNSB/BC	NRR/DSS/STSB/BC
NAME	THood	BAbeywickrama	SKrepel	VCusumano
DATE	02/11/2021	02/25/2021	02/23/2021	02/25/2021
OFFICE	NRR/DSS/SCPB/BC	NRR/DRA/APLA/BC	NRR/DRA/APLC/BC	NRR/DEX/EMIB/BC(A)
NAME	BWittick	RPascarelli	SRosenburg	ABuford
DATE	02/26/2021	01/21/2021	01/21/2021	03/01/2021
OFFICE	NRR/DEX/EEOB/BC	NRR/DEX/EICB/BC	OGC – NLO**	NRR/DORL/LPL2-2/BC
NAME	BTitus	MWaters	STurk	UShoop
DATE	02/22/2021	3/3/2021	03/22/2021	03/25/2021
OFFICE	NRR/DORL/LPL2-2/BC	NRR/DORL/LPL2-2/PM		
NAME	DWrona	THood		
DATE	04/02/2021	04/02/2021		

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